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INTRODUCTION TO NUCLEAR PROPULSION

Lecture 13 - REACTOR CONTROL

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INTRODUCTION TO NUCLEAR PROPULSION

Lecture 13 - REACTOR CONTROL

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[redacted]

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PREFACE

The following discussion is intended to point out some of the peculiarities of the reactor with respect to its control. No attempt is made to include any theory on servo-analysis inasmuch as many excellent references are available. Rather, the emphasis here has been to indicate reactor control characteristics and areas where reactor control is somewhat unique.

CONTENTS

1. INTRODUCTION

Before any process, machine or energy source can be employed usefully it must be made to conform to a required pattern of operation. Even the atomic bomb must be controlled to the extent that the operator has the prerogative of determining the time at which the sustained chain reaction is to be initiated, although the remaining sequence of events may be beyond his control.

The next two lectures will be aimed at indicating methods of control, mechanization and some of the control system characteristics required for reactor control.

In previous lectures you have covered much of the reactor technology required to understand the methods and techniques of controlling a nuclear reactor.

It would be well to review some of the areas which have the most influence on reactor control. Figure 1-1 will be used to illustrate the following reactor principles.

1.1 Sustained Chain Reaction

The sustained chain reaction is possible since a single neutron can cause a fission which in turn will produce ~ 2.5 neutrons. If the inventory of these fission produced neutrons can be regulated we then can control the number of fissions per second. We must contend with loss of these neutrons due to loss or leakage from the finite system, and non-fission capture of neutrons in the fuel, structure, and in the by-products of the reactor. Some of these methods of neutron inventory regulation are employed to control various reactors throughout the country.

1.2 Fission Energy

The energy release of a fissioning atom of ${}_{92}U^{235}$ results from the conversion of mass (binding energy) to kinetic to thermal energy. Each fission produces ~ 200 Mev, 3.2×10^{-11} watt-sec, or it requires 3.1×10^{10} fissions/second to produce 1 watt of power. With the reactor just critical, the loss of neutrons to non-fissioning processes is regulated so that only one fission neutron is available to induce another fission, (the remaining ~ 1.5 are lost by leakage or non-fissioning absorption.

The neutron population at any given time is a measure of the energy being released by the total core volume.

1.3 Delayed Neutrons

If it were not for delayed neutrons, control of a reactor would be impossible. Remember that $\sim 99.3\%$ of all neutrons are ejected from the fissioning atom within 10^{-14} seconds after fission while 0.7% are not produced till ~ 0.2 to ~ 80 seconds

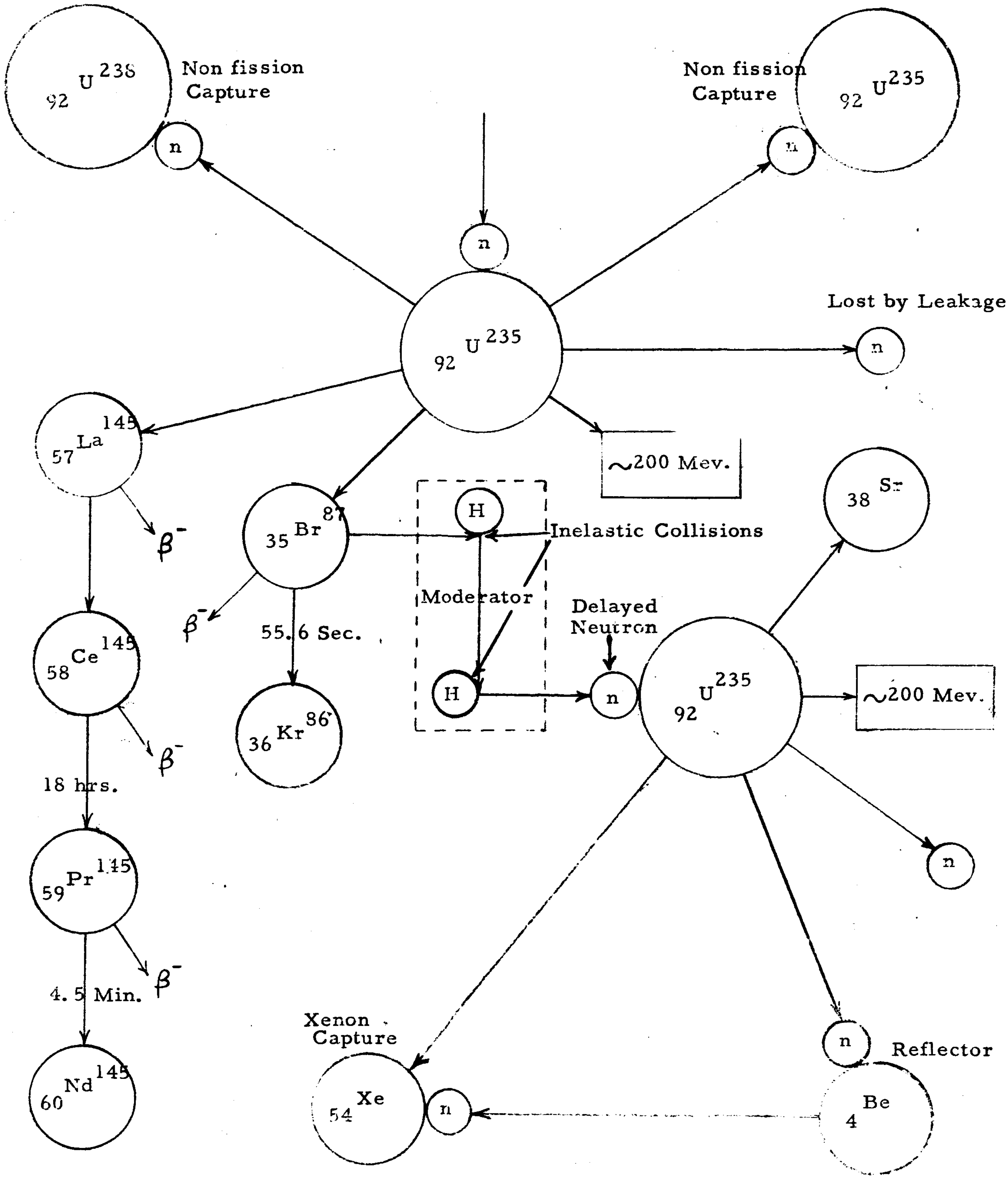


Figure 1-1
Chain Reaction

after fission has taken place. Since most of the neutrons are generated within 10^{-14} seconds and this second generation neutron can then induce another fission in $< 10^{-5}$ seconds the power can increase at a tremendous rate if the reactor is allowed to become supercritical, i.e. each fission results in >1 neutron available to cause another fission. No physical control system - which must move mass or inertial loads can be expected to cope with such a rate of power change. Therefore, we never let the reactor increase power on the prompt neutrons alone (prompt critical) but insure that the extra neutrons for power increases be limited to less than the delayed neutrons.

1.4 Fission Products

The fissioning ${}_{92}U^{235}$ atom breaks up into pairs of elements which spans a large spectrum in the periodic table. These fission products have various effects, some become non-fission absorbers Xe^{135}, while others become radioactive.

1.5 Temperature Effects

The effectiveness or availability of neutrons to cause fission is a function of the temperature of the core material since absorption cross sections (Σ_a) are effected. by kinetic energy of the various atoms, fuel, reflector, moderator, structure, etc.

1.6 Moderator and Reflector

Most of the original fission neutrons are produced with energies in the 0.1-5 Mev energy range, the delayed neutrons have energies of $\sim$0.25 $\sim$0.67 Mev. Also, the absorption cross-section of U^{235} is inversely proportional to the energy of the neutron and is optimized in the low-ev region (0.1 ev). Therefore, moderator material is added to the reactor core to slow down or remove most of the kinetic energy from the neutron. This energy removal is accomplished by inelastic atomic collisions which transfer the energy from the high velocity neutrons to the slower moving moderator atoms (carbon, hydrogen - low mass number atoms).

The loss of neutrons by leakage is attenuated by wrapping a blanket of material around the core which in effect returns a neutron back to the system when it is bombarded by neutrons from the core.

2. DEFINITION OF CONTROL

Any control system is a dual-function system designed to regulate operation and to protect equipment and personnel. The regulating function controls the reactor during normal operating conditions, while the protecting function overrides the regulating function during operating conditions which may result in damage to the installation. The relative importance and justifiable complexity of each function in the control system depend upon such factors as the application of the nuclear system, safety of the nuclear system, and hazards to surrounding areas.

The regulating function, using either open or closed loops, maintains a preselected controlled variable, or variables, in a fixed relationship to an input demand. The input to a regulating function may be an operator demand or may be computed by a larger external loop that measures and compares other control variables to determine the required demand. This demand signal is then compared to the control variable and any difference applied to initiate action of a control element to return this difference to null.

The automatic flux loop is a null-type regulating loop in which the neutron flux level is measured and compared to the input demand. Based upon the determined difference, the reactivity of the reactor is adjusted to force the reactor flux to assume the desired relationship to the demand. The degree to which the controlled variable matches and follows the demand signal is determined by several control system characteristics, including the type of system, system gain, and transient response of the system.

In some cases, the protective function may be combined with the regulating function to use the normal regulating elements in supplementary protective responses. Figure 2-2 shows a typical automatic-manual control system with supplementary protective responses.

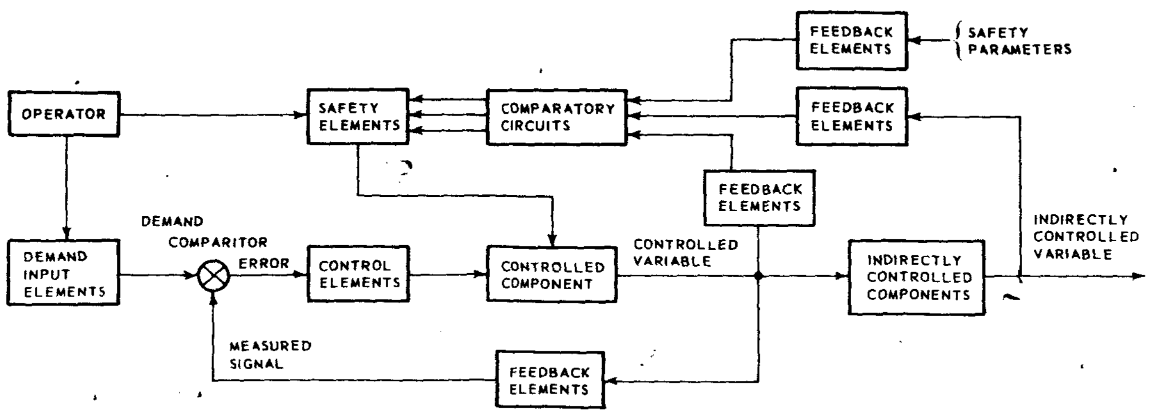


Fig. 2-2 – Automatic-manual regulating control with automatic protective system

The protective function, normally open-ended, initiates corrective action of a fixed magnitude and direction, when some measured parameter falls outside of a predetermined limit. The operator is an important factor in almost all protective systems. He monitors the various system parameters; in many cases, he is assisted by visual and/or audible alarms. Figure 2-3 illustrates a system in which the operator is the decision-making element in the protective system.

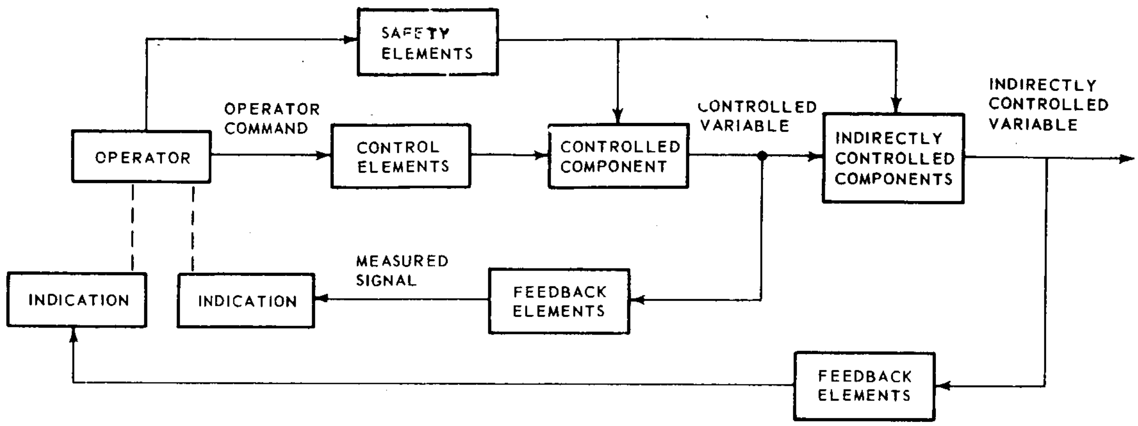


Fig. 2-3 – Manual regulating control with operator as protective monitor

The safety system may also take positive action by automatically limiting operation to some defined area, e.g., a maximum power level. In this case, the safety action overrides the automatic regulating system by preventing a measured parameter from exceeding a preset limit. This action is referred to as limiter action. The demand input can also be automatically changed by the safety function at a programmed rate or in fixed steps. Limiter action and changing the demand input requires that the automatic regulating system must be operating properly for limiter action or for a safety action, which operates on the input demand, to be effective. A more drastic action introduces a programmed step change or rate of change in some intermediate control parameter to produce a known effect on the measured parameter. Such action may include complete shutdown.

The final safety action should be completely independent of the other control systems and functions. Reactor scram exemplifies this type of action.

2.1 AUTOMATIC AND MANUAL CONTROL

When accurate control with good repeatable transient performance is desired, automatic control should be used. The complexity of the installation, the number of parameters to be monitored, expected life, duty cycle, and the effects of small perturbations in the control variables determine the extent that manual control may be applied.

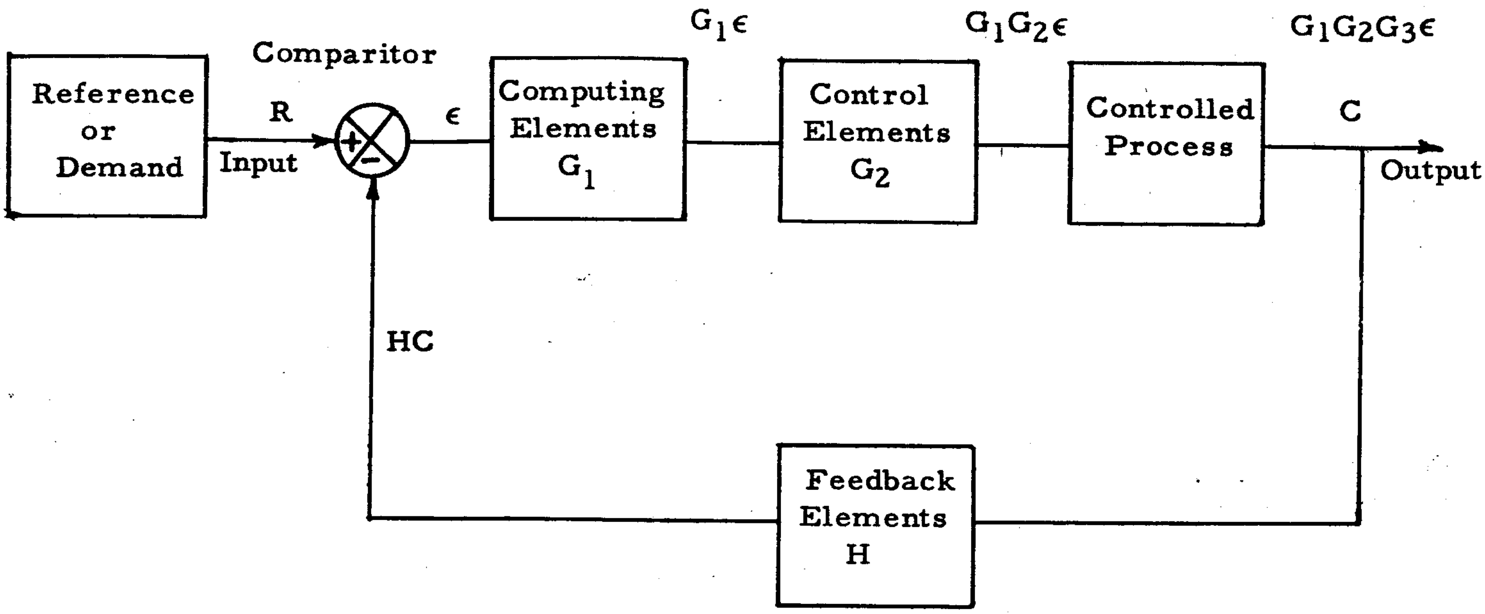
A central power station requires precise speed regulation of the alternators over long periods of time, necessitating an automatic control system. The educational-type reactor, however, normally does not require precise control. Reactor power is the primary control variable and operation normally continues over intervals of less than one hour. Hence, automatic control is of much less value to the installation.

Automatic control is normally closed loop in that the controlled parameter is measured and compared to the reference or demand to determine what corrective action, if any, should be taken. The following simple diagram, Figure 2.1-4, will serve to illustrate closed loop system operation. Any control system will include some or all of the indicated system parts.

2.1.1 AUTOMATIC CONTROL REQUIREMENTS

The control system requirements determine the degree of automation of the control system. The automatic control requirements include, as a minimum: controlled parameters, range of control, accuracy, and transient response.

1. Control Parameters - Possible control parameters on a nuclear system are: reactor period, reactor nuclear power, reactor thermal power, reactor discharge temperature, reactor discharge pressure, inlet or outlet temperature on the turbine, turbine shaft speed, inlet or outlet pressure on the turbine, or gas weight flow. Selection of the controlled variables may indirectly determine the control element. For example, reactivity control is used to regulate all reactor parameters, as well as turbine inlet temperature and pressure for a direct cycle.
2. Range of Control - The range of control is expressed from one level to another level, as from 200°F to 1200°F, or from 1 percent to 100 percent full power. In most applications, the control is not required to operate over more than a ten-to-one, or 1-decade change in the controlled variable. In the nuclear power control, the requirement may approach two decades of control (100 to 1) in the power range and 6 to 8 decades in startup range. A 2-decade control, using linear techniques approaches marginal component abilities. Any greater range should be discouraged for an efficient system. A single decade or 1-1/2 decades can be met with relative ease. Special non-linear techniques, used in the startup range, are described in section 4.4 .
3. Accuracy - Accuracy is normally stated as an operating tolerance about the demand level, and is expressed either in terms of percent of the operating point, percent of full power, or in absolute values: ±1 percent of operating point, ±0.5 percent of full power or ±10°F. The sensor is often the limiting factor for accuracy in many systems. Thus, the control system cannot be expected to maintain greater accuracy than the intelligence with which it operates. Assuming that the sensor fulfills the accuracy requirements, the remainder of the system can be designed to control with accuracies of ±1 percent over a decade of power range without imposing particularly difficult restrictions on the components. Accuracies of 0.5 or 0.1 percent, however, will require maximum component efficiency.
4. Transient Response - The transient response of a control system is a measure of the ability of the system to effect a programmed change in the control variable. This requirement may include acceleration and deceleration rates, step function responses, and system frequency response. The transient response of the system determines the



$C = G_1G_2G_3\epsilon$ Let $G = G_1G_2G_3$

$$
\begin{aligned}
C &= G\epsilon \\
\epsilon &= R - HC \\
C &= GR - GHC \\
C(1+GH) &= GR
\end{aligned}
$$

$$\frac{\text{Output}}{\text{Input}} = \frac{C}{R} = \frac{G}{1+GH}$$

Thus increasing the forward Loop Gain, G, will force the Output, C, to agree more closely with the input, R.

Figure 2.1-4
Typical Control System - Block Diagram

maximum rate for a change in the control parameter, and the ability of the control system to damp out extraneous perturbations. The system gain and system time constant are partially determined by the required frequency response of the system.

2.2 MANUAL CONTROL

In some applications, control is accomplished mainly by manual controls with the operator performing the regulating and monitoring function. Other installations use the operator only to monitor the control system, serving as a backup in case of failure in the automatic system. Practically all nuclear systems, regardless of the degree of automation, include some means for the operator to manipulate the reactivity actuators and to initiate various safety actions.

The operator may control the actuators by means of closed-position loops, or he may directly control individual actuators, or groups of actuators in open-loop systems, by controlling the drive motor or servovalve. The open-loop capability increases the reliability of the manual control since the only components needed for control are the actuators and the command-transmitting circuitry. This ability to control reactivity permits operation to continue during practically all control system malfunctions, as long as the operator has feedback information from his control sensors and maintains control of the actuators.

The operator can normally override the automatic system whenever he determines that existing conditions are not in the best interest for continued operation.

2.2.1 MANUAL CONTROL REQUIREMENTS

In a manual control system, the operator or operators control accuracy and rate of response. So long as the control elements do not limit the ability of the operator to control the power plant, the absolute response of the control loop cannot be accurately predicted. Since the operator contributes intelligence and logic to the control system, reflecting past history and operating experience, accuracy and response are normally not included in the requirements for manual control.

1. Control Parameters - The manual-automatic control philosophy determines the parameters to be controlled manually. Normally, the operator has some manual control as a backup or override of the automatic system. The manual control parameters may duplicate parameters to be automatically controlled; in some cases, however, additional parameters are included. The nuclear power plant which has only minimum manual control normally permits control of reactor power by providing manual reactivity control. When automatic control is used, manual control of the control element should normally be provided as a backup in case of malfunction in the automatic loop.
2. Range of Control - Manipulation of the control variable by the operator may be limited. In some cases, this limitation may necessitate automatic limits; in other cases, limits are imposed by documented operating instructions. The operator may be limited to only one direction of control, or he may be given the perrogative of overriding or bypassing the automatic control whenever he determines such action to be necessary for safe operation.

2.3 SAFETY SYSTEM

The greater application of nuclear reactors and the increasing complexity of the larger nuclear systems have tended to increase the emphasis on protective systems. As the design technology has advanced, power systems have increased in size, cost, and complexity, partially because of the trend toward automation and the requirement for maximum efficiency of operation. The investment, coupled with the complexity of the installation and its automatic control system, determines the need for protection. For system operation at or near peak efficiency, the power plant components must usually function near their operating limits. The accuracy and time response required to provide protection of many parameters, under these circumstances, is beyond the capabilities of the operator.

2.3.1 SAFETY REQUIREMENTS

These requirements, and the resulting safety system design, are determined to a great extent, by the control and installation operating philosophies. Safety systems normally deal with at least three factors: malfunctions, safety actions, and safety parameters.

1. Malfunctions - The safety requirements of a particular system are based, in part, upon the types of malfunctions against which the safety system is expected to guard. Possible malfunctions include: control system or component failure, operator error, natural causes, such as earthquake, facility failures, accessory failures, external perturbations, e.g., a rammed ship, and internal perturbations, e.g., a fuel shift within the reactor core.
2. Safety Actions - Safety requirements indicate the various actions that may be safely applied in a particular situation. Under certain conditions, a slow reduction of power may result in aggravating an undesirable situation. If a decrease in nuclear power results in a corresponding but larger decrease in coolant flow to the reactor core (assuming that the coolant pump receives power directly from the reactor coolant and that the pump has a steep flow versus power characteristic), a power reduction to correct an overtemperature would only increase the overtemperature.
3. Parameters - The parameters that are most indicative of damage or of impending damage must be considered in the safety-system design. Abnormally high reactor flux is not necessarily an indication of overtemperature; however, it is an indication of impending overtemperature. Sufficient reactor power, over an extended period, produces sufficient heat energy to raise the fuel element temperatures to the melting point. Reactor power, then, is an anticipatory signal indicating an impending overtemperature. One signal can be more valuable than another signal as a safety parameter although the signals are ultimately an indication of the same condition. Parameters and conditions which should be considered in the safety system design include: pressures, temperatures, shaft speeds, reactor period, reactor power, actuator and valve positions. Other factors necessitating careful design for safe operation include: control system continuity, radiation levels, electrical, hydraulic, and pneumatic power supplies, personnel doors, interlocks, and supporting equipment. Consideration of safety parameters should also include the maximum and minimum values for safe, continuous operation.

3.0 REACTOR CONTROL CHARACTERISTICS

Every reactor has some peculiarities which require special consideration. A reactor is capable of operating over a very broad power range; it has a gain which is a function of the power level; and it will behave as an integrator if there are no temperature coefficients. In addition, a reactor may contaminate its environment, necessitating remote location or adequate shielding for both the operating personnel and the control system components.

The ability of the reactor to operate over a range of power levels of approximately ten decades, requires particular emphasis on safe operation because the reactor exhibits the same characteristics, neglecting the temperature effects, in the low power regions as in the upper power range. The low-level signals, with magnitudes approximately equal to induced electrical noise, accentuate the measurement problems during low-level operation around 10^{-4} to 10^{-8} percent full power (FP).

Because the reactor is generating insufficient power to satisfy the needs of the power converter when reactor power is below 1-10 percent FP, the power levels below 1 percent FP are unimportant to power-range operation. Thus, the control requirements for the operating levels or power range are quite different from the requirements for control in the low power range. The power-range control requirements include accuracy, rate of response, and safety, while the important requirement in the low-power levels is safe operation.

Because of the two different requirements, two and sometimes three systems are built to span the ten decades of power and various ranges, including power range, ~1 to 100 percent FP; intermediate range, 10^{-4} percent to ~1 percent FP; and source range, 10^{-8} percent to 10^{-4} percent FP. The 10^{-8} to ~1 percent FP range may be covered with a startup system.

The combination of the variable gain and the integrating features of the reactor result in an exponential integrator. The power can be represented by the equation:

$$\phi = \phi_0 e^{\frac{t}{\tau}} \tag{1}$$

where

ϕ = nuclear power at time t
ϕ_0 = nuclear power at t = 0
$e \cong 2.718$
t = time
τ = the reactor period

Three possible methods of linearizing the reactor gain are shown in section 4.2 Equation (15).

Reactor operation is also affected by the past operating history of the reactor. These affects result in changes in the multiplication factor or reactivity of the reactor. The next few paragraphs will describe some of the reactivity changes, causes, and results.

3.1 REACTIVITY COEFFICIENTS

Temperature coefficient of reactivity, xenon generation, and fuel depletion are examples of predictable natural phenomena which affect a reactor during operation. The temperature coefficient affects the dynamic control system requirements, while the xenon generation and fuel depletion require the provision of some means of introducing positive reactivity into the reactor over a long interval. Introduction of +10 to +15 percent Δk into the core may be necessary to override maximum shutdown xenon.[1]

3.1.1 Temperature Coefficient

A reactor operates at temperatures ranging from room to maximum operating temperature. A change in reactivity can result from the temperature change. The rate of change of reactivity with change of temperature is defined as the temperature coefficient of reactivity.

The temperature effect on reactivity is not always a fixed or linear relationship. A negative temperature coefficient adds negative reactivity to the reactor as the temperature increases. Thus, there is an inherent tendency for the reactor to be self-limiting as the temperature rises. The effect of the positive temperature coefficient is to increase reactivity as the temperature increases which tends to increase the rate of power rise.

While the reactor with a positive temperature coefficient has no inherent tendency to be self-limiting, it can be controlled and operated safely. A reactor with a positive temperature coefficient imposes greater demands on the control system than the same reactor with a negative coefficient. Emphasis must be placed, particularly during initial design phase, upon system and component reliability, safety response time constants, and operating procedures.

The temperature effect on reactivity consists of: the density effect due to the change in density of the reactor materials and the change in volume of the reactor with temperature; the nuclear effect resulting from the variation in the thermal flux spectrum and the Doppler broadening with increased temperature; and the effect caused by the dissociation of hydrides at high temperature. In a solid-moderated reactor, the effect of the change in density and volume is of such small magnitude, as compared to the thermal flux spectrum shift, that it may be neglected.[1] The effect of the dissociation of hydride is also neglected since it is presumed that normal operating temperatures will be maintained below the dissociation temperature of the moderator. An inadvertent overtemperature of a solid moderator that might result in hydride dissociation becomes a negative reactivity coefficient and thus tends to assist the control system in limiting the overtemperature. This does not affect the dynamic requirements of the control system but could necessitate additional shim reactivity.

The thermal energy of the reactor flux at $68^{o}F$ extends from 0 to approximately 0.06 ev. As the reactor temperature increases, the energy band of the neutrons widens or increases in energy, changing the energy distribution of the flux in the thermal energy region. Since the microscopic cross sections of the reactor materials are dependent upon the energy of the neutron flux, a change in the energy distribution changes the number of scattering, absorbing, and fissioning events occurring per unit time in the thermal energy region. The first effect of increasing the temperature is the increasing of the energy of the moderating nuclei, tending to shift the entire flux distribution toward higher energies. Other factors which also affect the thermal flux distribution include the type of moderator, poison, and leakage.

The Doppler[2] effect is a broadening of the resonance region due to an increase in the energy of the fuel nuclei with temperature. As the fuel temperature increases, the effective fission cross section of U^{235} is increased, resulting in a positive reactivity effect. This positive temperature coefficient is somewhat lessened by an increase in the nonfissioning absorption cross section of the U^{235} nuclei. The Doppler broadening can also result in a negative temperature coefficient if the nonfission cross section increases faster than the fission-capture cross section as a function of temperature. In intermediate reactors, the Doppler effect may be negative because of resonance regions with a high nonfission capture to fission capture ratio. Thus the Doppler effect can normally be ignored except in the fast reactor. However even in the fast reactor, the admixture of sufficient U^{238} with the U^{235} can adjust the Doppler effect. The Doppler broadening effect is often neglected in the thermal and epithermal reactors. Therefore, the only temperature coefficient to be considered is due to the variation of the thermal flux spectrum.

The total temperature effect on reactivity includes variations in the thermal flux spectrum on the fuel elements, the moderator, and the reflector. In metallic-fueled reactors, the fuel elements, moderator, and reflector are three separate components of the reactor and each demonstrates its own unique coefficient and time constant. In metallic reactors, the temperature coefficient is dominated by the effect of the moderator. Since the major moderating or slowing down of the fission neutrons occurs in the moderator, the energy of the nuclei of the moderator tends to determine the energy spectrum of the reactor. The moderator thermal time constant is considerably longer than that of the fuel because of the greater thermal capacity of the moderator. Thus, the moderator determines the time constant and the magnitude of the dominant temperature coefficient. However, because of the long moderator thermal time constant, the temperature coefficient of the fuel element is important in determining the dynamic requirements for the control system. Therefore, during transient conditions, the control system must include sufficient dynamic response capability to meet the demands of the fuel element temperature coefficient. The moderator and reflector thermal time constants are usually greater than the fuel element time constant; consequently, they are of less importance to the dynamic control requirements. These thermal reactivity coefficients are, however, important factors in determining the reactivity of the shim system.

In the ceramic reactor, the moderator and the fuel materials form a single component with a single time constant and effect. The result is a lengthening of the fuel element time constant and a decreasing of the moderator time constant. The core temperature coefficient remains negative and is the predominant time constant for determining dynamic control requirements. Figure3.1-5shows a typical curve of the combined fuel and moderator temperature coefficient in a ceramic core. This particular curve pertains to a core with a temperature coefficient of approximately -8.8×10^{-4} percent $\Delta k/^{\circ}F$. The curve in Figure3.1-6shows the results of adding the temperature effects of the reflector to the core. The temperature coefficient reverses at approximately $600^{\circ}F$. The curve shown in Figure3.1-5represents the transient temperature coefficient to which the control system must respond, while Figure3.1-6represents the combined temperature coefficients and indicates the steady state reactivity requirements placed upon the control system. Although the temperature coefficient during the transient condition is negative, relieving the dynamic response of the control system to some degree, there is a temperature region in which a positive temperature coefficient exists during steady state conditions. This positive temperature coefficient imposes some restriction on the control system reliability and operating procedures.

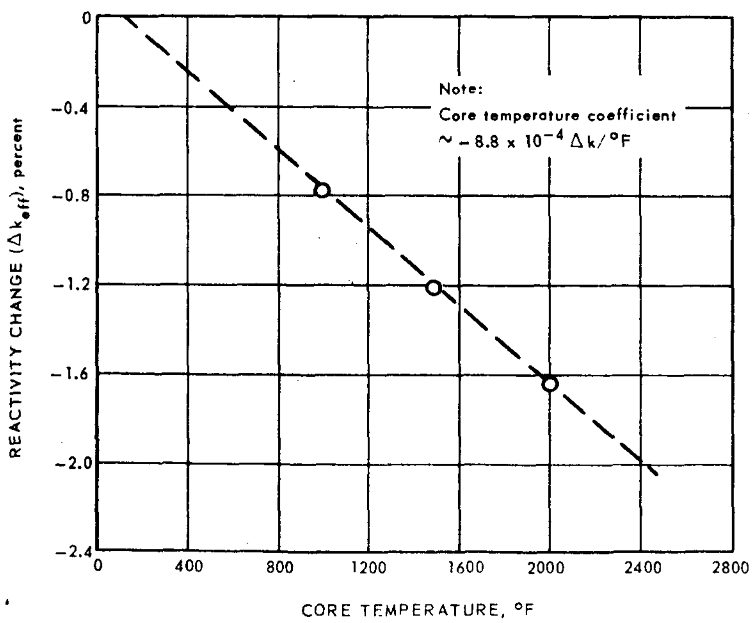


Fig. 3.1-5 – Reactivity change versus temperature for a typical ceramic core (constant reflector temperature)

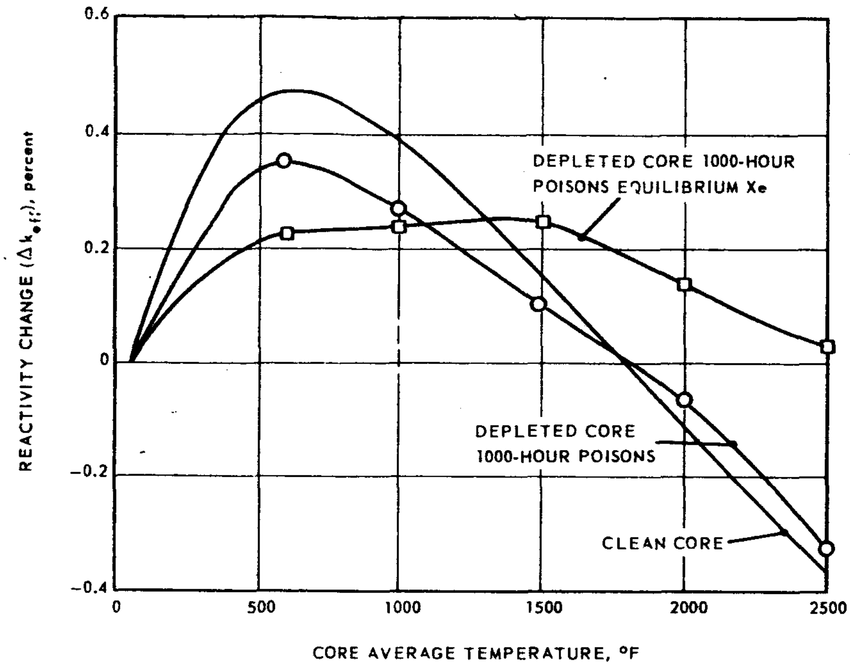


Fig. 3.1-6 – Reactivity changes versus combined core and reflector temperature for a reactor with a ceramic core

The effect of temperature coefficients[3] is illustrated in the two curves shown in Figure 3.1-7. These two curves indicate the power excursion for a typical startup accident. Curve A assumes a negative linear temperature coefficient of -2 x 10^{-4} percent $\Delta k/^{0}F$ while curve B assumes a core temperature effect similar to the combined core and reflector temperature effect shown in Figure 3.1-6. Curve B represents a purely hypothetical case since the reflector was combined with the reactor core and consequently is assumed to have the same thermal time constant as the fuel and moderator. Curve B shows reactor power beginning to decrease before introduction of scram reactivity at t = 10.365 seconds. The power decreases because the temperature coefficient becomes negative as the temperature rises above 600^{0}F. When the reactor temperature reaches approximately 2300^{0}F, sufficient negative reactivity has been introduced to cause the reactor to become subcritical.

To obtain the data for the curves, the following assumptions were made during the programming of the digital computer:

1. **All controls and safeties had failed except the high level nuclear power safety. The control rods were withdrawing at maximum rate of +4 percent Δk/minute.**
2. **The reactor became critical at 1 watt.**
3. **No heat had been removed from the reactor which was initially at room temperature.**
4. **The delay time between the operation of the safety trip at 125 percent full power and initial insertion of negative reactivity was 130 milliseconds. Reactivity was inserted as linear function of time with - 2 percent Δk inserted in 300 milliseconds.**

The method of simulating the temperature coefficient is quite simple if the reactor simulator includes fuel element temperature or reactor coolant outlet temperature. Thus, the effect caused by the fuel elements, and, in case of a homogeneous reactor, by the moderator, is a negative feedback of reactivity as a function of temperature. Both the moderator, if fuel and moderator are separate, and the reflector reactivity feedback require a gain. This gain is composed of the temperature coefficient and the gain term for

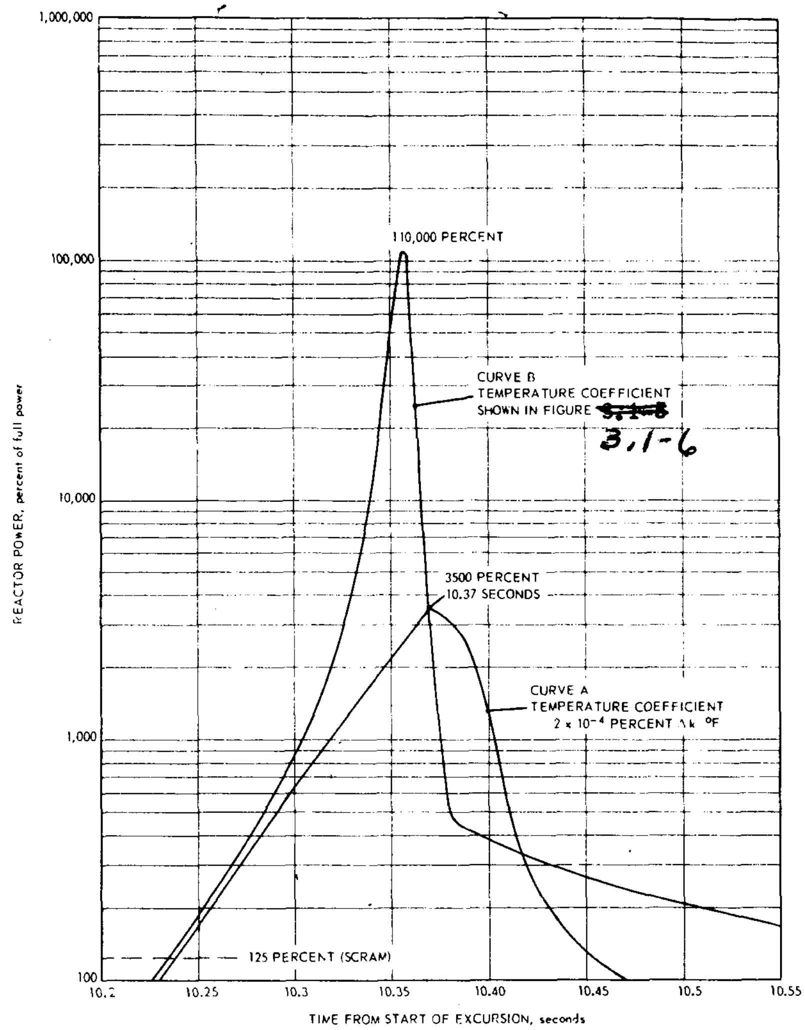


Fig. 3.1-7 – Reactor power (in percent full power) versus time for a startup accident

nuclear power to moderator and reflector temperature, with a time constant due to the lag between the nuclear power generated in the fuel and the moderator and reflector temperature. If the temperature coefficient is nonlinear, inclusion of a function generator may be necessary.

3.1.2 Xenon Generation

Xenon has a very pronounced effect upon the reactor operation, but because of the long time constants involved, is not considered important to the dynamics of a reactor control system. The buildup of xenon, a very slow process requiring hours to reach equilibrium, necessitates only a slow compensating control to cancel its effects. The rate of xenon burnup, however, may require investigation. When the reactor is shut down, the xenon level builds up to a peak value, and then decays slowly to some equilibrium level. If the reactor is restarted with the xenon level at or near peak value, the xenon will be burned or depleted at a much faster rate than the normal decay. This depletion of xenon increases the reactivity of the reactor and must be compensated for by negative reactivity to maintain the demanded power level. While the xenon burnup rate is much faster than the buildup or production rate, and should be investigated as a possible lower limit on reactivity insertion rate of the control system, the normal operating requirements are usually more demanding of the control system than is the xenon burnup rate. The slow buildup and rapid burnup of xenon[4] are shown in Figure 3.1-8 As shown on the broken line curve, the xenon buildup rate and the burnup rate are approximately the same when the power first decreases and then increases. The increase in xenon results when the power is decreased from 100 percent to 50 percent for one hour. After the xenon starts its initial rise, but long before it begins to approach its equilibrium value, the power is increased to 160 percent and the xenon begins to drop. In the xenon buildup, negative reactivity is introduced into the reactor at approximately 0.5 percent Δk/hour which is within the capability of any control system. The burnup results in the same rate of positive reactivity addition. A faster rate of burnup can be obtained if the reactor is started and brought to power at the maximum shutdown xenon level of 47 hours.

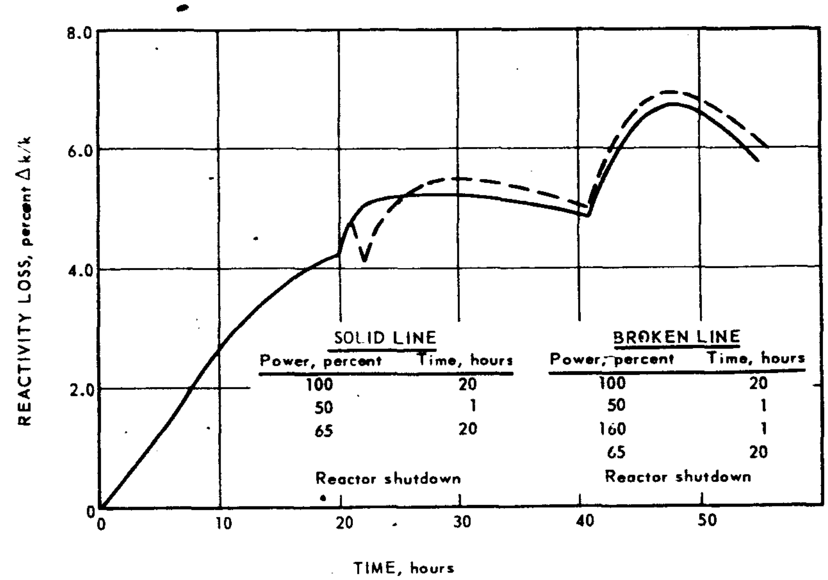


Fig. 3.1-8 – Xenon-135 effect on reactivity versus time

3.1.3 Fuel Depletion

As fuel is burned or consumed in the reactor, there is a decrease in the reactivity. This reactivity decrease, like the xenon production, occurs so slowly that it has no effect on the dynamic requirements for a control system. The rate of fuel burnup is proportional to the power level and time at power, given in megawatt hours. A typical rate of burnup, -10^{-5} percent Δk/megawatt[4] hour or 10^{-3} percent Δk/hour for a reactor operated at 100 mw, is insignificant for dynamic control.

Both xenon and fuel burnup result in very slow negative reactivity effects of a relatively large magnitude especially when the effects of peak shutdown is concerned. Since this negative reactivity may become greater than 15 percent Δk for a long life reactor, some method of inserting positive reactivity is required. The shim controls are designed to provide the required positive reactivity.

4.0 CONTROL PROCESS

Control of the reactor consists of regulating the efficiency of the fission neutron to fission capture process. Regulation of the neutron production may be accomplished by: (1) removing neutrons from the reaction by adding parasitic absorbers; (2) reducing the fissioning material in the process by removing fuel; (3) allowing more neutrons to escape from the reaction by removing reflector material; or (4) decreasing the probability of fission capture by removing moderator material. The choice of control method is determined by the energy spectrum of the reactor. If the reactor neutron energy levels are in the thermal range, absorber materials are often used. The capture cross section of boron is proportional to the reciprocal of the neutron velocity: $\sigma \approx \frac{1}{v}$. Therefore, because of the high energy levels of the neutrons, absorber control is normally not used in fast reactors.

Since the reactor is an integrator, only a small amount of control should be needed to operate the reactor at any power level. The minumum controlled reactivity is determined only by the desired rate of power increase, with the maximum reactivity set by the maximum rate of power rise for safe operation. However, the reactor has inherent characteristics which also affect the fissioning process efficiency. Some of these characteristics result from the instantaneous condition of the reactor, while others result from its past operations. The characteristics that are determined by the instantaneous condition of the reactor affect the requirements that apply to the transient performance of the automatic control system. The characteristics resulting from past operation of the reactor are slow and require a low rate of compensating reactivity addition. Fortunately, the fast responding characteristic is small in magnitude, while the slower response has a much larger magnitude.

Due to the difference in magnitude and response rates of these inherent characteristics, requirements for two different types of reactivity control have sometimes resulted. One reactivity subsystem, with an amount of reactivity usually less than β effective, is capable of changing the reactivity at a rapid rate but has a limited total integrated reactivity change, Δk. This system performs the regulating function for the control system and thus determines the dynamic response of the control system. A second reactivity subsystem is much slower but can contain more than twenty times as much reactivity as the dynamic or regulating elements. The slow system is often referred to as the shim system. In some automatic controls the shim control elements have been electrically slaved to the regulating elements. As the regulating elements approach their control limit, the shims are actuated to assist the regulating reactivity subsystem, thus preventing the regulating elements from becoming saturated. This type of control can be further extended, using the shims to maintain the regulating elements at or near some predetermined referenced position at all times, except when the control requirements are for reactivity rates which exceed the capabilities of the shim system.

Regardless of which parameter is the controlled parameter, one of the manipulated variables will be nuclear power or its derivative. Reactor control must, therefore, include some method of regulating the neutron inventory. The method assumed throughout this section involves the placing of long thin rods of absorber material along axes parallel to the axis of the core itself. Insertion of the absorber or poison, $-\Delta k$, depresses power, while removing or withdrawing the poison increases the reactivity. The absorbers can be replaced by fuel or moderator material and thus be used for control. The effect will be reversed since an inserted fuel rod or moderator rod causes an increase in reactivity and reactor power. Reflector control is usually accomplished by placing drums at the periphery of the core. One half of the cylinder is covered with a neutron reflecting material that attenuates the escape of neutrons while the other half of the drum is coated with a neutron absorber. By regulating the surface of the drum exposed to the core, neutrons may be either reflected back into the system or absorbed. The drum can be positioned to produce any desired effect between these two extremes. With a drum control, the control element displacement is measured in Δk per radian, while the rod-type actuator longitudinal motion is measured in Δk per inch.

Power range control can be subdivided into reactivity control, neutron flux control, and temperature control.

4.1 REACTIVITY CONTROL

Performance requirements for reactivity control are based on a number of factors, including: the type of neutron control involved, whether absorber or reflector; the number of control elements, the location of the elements, the amount of travel, power profile requirements, and the total reactivity.

The rates of change of Δk to meet the overall performance requirements must be determined. This will, in turn, determine the need for one or two types of actuators. If satisfactory control can be maintained by Δk rates, approximating 1 percent per minute, only shim actuators are required. Shims are normally used to compensate for reactivity changes such as xenon buildup, fuel depletion, and the steady state temperature coefficients; they can also be used, however, for dynamic control if the required reactivity rates are sufficiently low. The Δk rate limitation is placed upon the shims since the shims will contain much more than enough reactivity to make the reactor prompt critical, thus generating periods on the order of 0.1 second, depending to some degree on the neutron thermal spectrum. A period is the time for power to change by a factor of e, 2.718, or to change from 100 megawatts to 271.8 megawatts. A U^{235} fueled reactor requires ~+0.75 percent Δk to become prompt critical. The Δk required for the reactor to become prompt critical is determined by β effective. Most shim systems will control 8 to 15 percent Δk. Therefore shims are rate-limited to reduce the possibility of such short periods being generated.

In order to circumvent this rate limitation on large Δk capabilities, a second type of actuator is used which has a high rate but a total Δk available for control limited to less than β effective, ~0.7 percent Δk. Assuming that the slow shims are not capable of the required regulating accuracy or transient response, faster regulating controls must be used. It is necessary to determine the optimum Δk rate needed for control system to meet the performance requirements. This rate is likely to be approximately 5 to 7 percent Δk per minute.

Actuator mechanization considerations indicate the desirability of selecting the minimum Δk rate that permits the control system to meet the transient requirements. The actuator represents a gain and several time constants to the control system; the time delays in the actuators usually comprise the dominant time constant in the reactor flux loop. The actuators also introduce nonlinearities into the loop in the form of saturation, Δk limited especially in the regulating actuators; velocity saturation, Δk rate limits; backlash and deadband. The system tolerance to backlash and deadband dictates the allowable tolerances for the actuator. The tolerance requirement should be based on a balance between system performance and hardware design problems. The backlash in a gear train can be reduced by the use of precision gears, but only with an increase in cost. The effect of deadband can be reduced by adding a low-amplitude fixed frequency signal to an electric motor input winding or by underlapping a spool valve. When regulating actuators are used, the backlash and time constant requirements on the shim actuators can be relaxed since the regulating actuators furnish the accuracy characteristic.

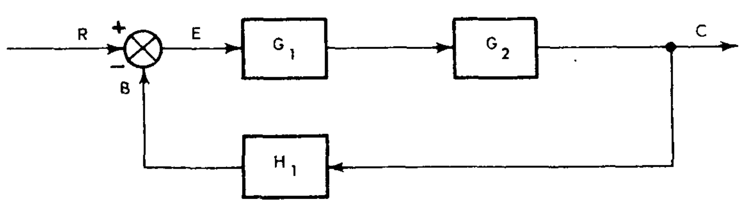
The actuators may be operated open or closed loop depending on the accuracy requirement for position synchronization between similar actuators. Open loop is generally preferred since it is less complex and requires less equipment. For shim actuators, relatively good position synchronization may be maintained by using synchronous motor drives. It may be desirable for the operator to maintain the actuators synchronized and to use ordinary induction motors.

There are, however, several reasons why closed position loops may be desirable for the regulating actuators. It is desirable to have more than one regulating rod to increase reliability and the required reactivity may not always be obtained in one element. Because several open-loop actuators operating in parallel may compensate for each other until one or more actuators reaches extreme limit of travel, closed position loops become more important. Closed position loops would force the actuators to assume nearly identical positions. With a shim control used to maintain the regulating actuators at their neutral position during steady state conditions, the transient response of the reactor-power control loop will be much more consistent. If an integrator is required to maintain accurate flux control, however, the open-loop actuators serve as an integrator to the system and will increase the system accuracy.

Shim actuators may be designed for manual or automatic operation.

In manual operation, the operator positions the shims to allow the regulating actuators to remain in a neutral position. During automatic operation, the shims are positioned by a command from the regulating actuators or from the flux error signal. When the regulating actuators must move some predetermined distance from neutral position, the shims move to allow the dynamics to return to neutral position. When closed-position loops are used for the regulating actuators, the magnitude of error may be used to initiate the shim command.

A typical closed loop for a regulating actuator is shown in Figure 4.1-9 The illustrated loop assumes a hydraulic actuator, hydraulic servovalve, electronic amplifier, and a linear voltage feedback.[5]



R = reference input, volts, ac
B = feedback signal, volts, ac
E = loop error, volts, ac
C = controlled variable (rod position), inches
G_1 = transfer function of error amplifier, milliamperes/volt
G_2 = transfer function of valve and actuator, inches/milliamperes
H_1 = transfer function of position sensor, volts/inch

Fig. 4.1-9 - Block diagram of regulating rod position loop

The transfer function for the actuator and servovalve is expressed:

$$G = \frac{K}{s\left(\frac{s^2}{\omega_0^2} + \frac{2\xi s}{\omega_0} + 1\right)} \quad \frac{\text{in.}}{\text{ma}} \tag{2}$$

where

ω_0 = undamped natural frequency, radians/second
ξ = damping coefficient
K = gain, inches/milliamperes
s = Laplace transform variable

To simplify the mathematics the following approximation is used:

$$G_2 = \frac{32}{s\left(1 + \frac{s}{80}\right)\left(1 + \frac{s}{240}\right)} \quad \frac{\text{in.}}{\text{ma}} \tag{3}$$

and (in Figure 4.1-9)

G_1 = 1.0 milliamperes/volt
H_1 = 2.0 volts/inch

An electronic amplifier is represented by G_1 and is assumed to contain no time lags since the normal time constants for an electronic amplifier are much smaller than the inherent time constants of an actuator.

The Bode attenuation-frequency diagram is used to calculate the closed loop response. Root-Locus, Nichols Charts or other methods could have been used. By plotting G_1G_2, and $1/H_1$ as two curves on the same graph and using the lower curve throughout the frequency spectrum to define the closed-loop response,

$$\frac{C}{R} \approx \frac{0.5}{\left(1 + \frac{s}{75}\right)^2 \left(1 + \frac{s}{240}\right)} \quad \frac{\text{in.}}{\text{volt}} \tag{4}$$

The closed-loop transfer function was derived mathematically and showed only slight loss of accuracy in the above approximation.

The following equation is based on the assumption that the actuator is normally maintained in a one-half withdrawn position with a total control element of 0.7 percent Δk for full travel.

$$G_3 = G_{\Delta k} = 3.5 \times 10^{-4} \frac{\Delta k}{\text{in.}} \tag{5}$$

This value of $G_{\Delta k}$, if stepped into a critical reactor, places the reactor on a steady-state period of approximately ten seconds. The transfer function of the regulating-actuator position loop becomes:

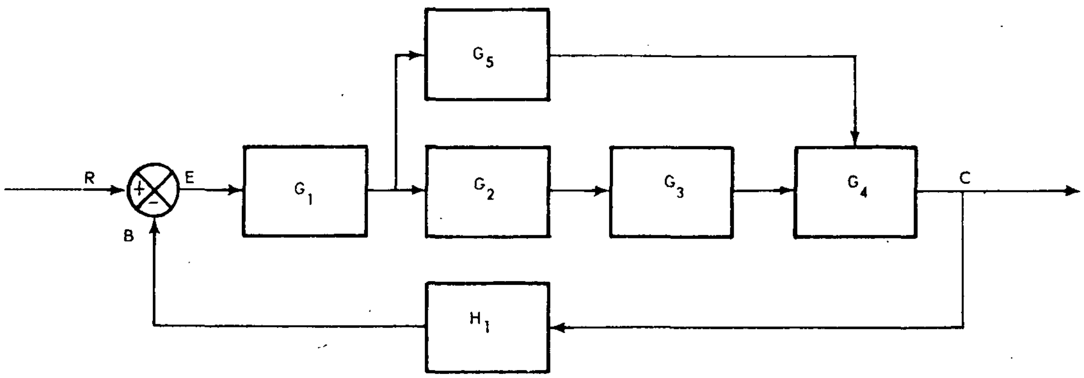
$$\frac{C'}{R} = \frac{0.5\,(3.5 \times 10^{-4})}{\left(1 + \frac{s}{75}\right)^2 \left(1 + \frac{s}{240}\right)} \quad \frac{\Delta k}{\text{in.}} \tag{6}$$

4.2 AUTOMATIC FLUX CONTROL

Practically all reactor control systems include either a manual or automatic flux control. Flux control is normally used as an internal loop when reactor outlet temperature is the control parameter and reactor neutron power is only of indirect interest. The flux loop is included inside the larger loop to provide the following:

1. A method of stabilization due to fast time response
2. A control parameter at the low power levels where insufficient thermal power may be measured, thus providing a power-holding capability at the upper power limit of the startup system
3. A backup control in case of a failure in the outer loops
4. Anticipatory limiting action before parameters reach destructive levels.

An automatic flux loop normally encloses the reactivity loop, similar to the one described in section 4.7, the automatic flux loop compares the measured flux to the flux loop demand and operates on the difference to calculate a reactivity loop demand.[5] A typical flux-control block diagram is shown in Figure 4.2-10.



R = reference input, volts, dc
B = feedback signal, volts, dc
E = loop error, volts, dc
C = controlled variable (neutron flux, percent of full power), % φ
G_1 = transfer function of stabilizing elements, volts/volt
G_2 = transformation of three paralleled position loops, inches/volt
G_3 = inches to Δk conversion for dynamic rods, Δk/inch
G_4 = reactor, $\%\varphi/\Delta k$
G_5 = shim circuit
H_1 = transfer function of ion chamber, volts/$\%\varphi$

Fig.4.2-10 Functional block diagram of flux loop

The dynamic control rods are operated in three parallel, closed position loops to maintain greater reliability and rod synchronization. The regulating-rod position loop-transfer function is the same as that derived in section 4.1, Equation (6).

The accuracy of the flux loop may be improved by using an integrator as the loop-error amplifier. No serious stability problem is introduced since the regulating actuators are enclosed in position loops. The typical transfer function of the error amplifier is:

$$G_1 = \frac{K_1}{s} \qquad (7)$$

A reactor transfer function may be derived by applying the principle of small increments to the reactor kinetic equations. An approximate expression of the reactor transfer function is:

$$G_\phi(s) = \frac{\Delta\%\phi}{\Delta(\Delta k)} = \frac{\%\phi_0}{s\left[l^* + \sum_{i=1}^{6} \frac{\beta_i}{\lambda_i + s}\right]} \qquad (8)$$

where:

l^* = effective neutron generation time
β_i = effective fraction of delayed neutrons from the ith group
λ_i = decay constant of the ith group of delayed neutron precursors

Because this equation does not readily lend itself to servoanalysis, the expression has been evaluated by digital computation and a family of gain and phase angle curves obtained for various values of l^*. These curves are shown in Figures 4.2-11 and 4.2-12.

By curve fitting in Fig. 4.2-11 the following approximate transfer function is obtained for the reactor:

$$G_4 = G_\phi(s) = \frac{10\%\phi_0 \left(1 + \frac{s}{0.027}\right)\left(1 + \frac{s}{0.4}\right)}{s\left(1 + \frac{s}{0.13}\right)\left(1 + \frac{s}{200}\right)} \frac{\Delta\%\phi}{\Delta(\Delta k)} \tag{9}$$

The open loop transfer function for the position loop, reactor and integrator is:

$$GH(s) = K_1 \frac{(0.5)\,(3.5 \times 10^{-4})\,(10)\,(\phi\%)\left(1 + \frac{s}{0.027}\right)\left(1 + \frac{s}{0.4}\right)}{s^2\left(1 + \frac{s}{0.13}\right)\left(1 + \frac{s}{75}\right)^2\left(1 + \frac{s}{200}\right)\left(1 + \frac{s}{240}\right)} \tag{10}$$

where an ion chamber is used as the feedback sensor. The ion chamber current signal may be changed to a voltage signal by measuring the drop across a fixed resistor to give the transfer function:

$$H_1 = \frac{1 \text{ volt}}{\%\phi} \tag{11}$$

The transfer function has a low frequency gain that is dependent upon the reactor power operating level, percent of ϕ. This variation in the open-loop, transfer-function gain is undesirable since transient response characteristics change as the operating level changes, making control loop stabilization difficult.

G_1 or H_1 must include a term inversely proportional to ϕ percent, so that the open loop transfer function is linearized and independent of operating level. This requirement may be met through:

1. Gain variation by demand position. The flux-error amplifier gain may be continuously varied as a function of the flux demand setting, as shown in Figure 4.2-13.

   Summarizing the currents at the amplifier inputs gives:

$$\frac{V_D + V_{fb}}{R_1} = \frac{V_e}{R_2}\left(\frac{\beta R_5 + R_6}{R_5 + R_6}\right) \qquad \text{where } R_2 >> R_5 + R_6$$

$$V_D = V_{Ref}\left(\frac{\alpha R_3 + R_4}{R_3 + R_4}\right)$$

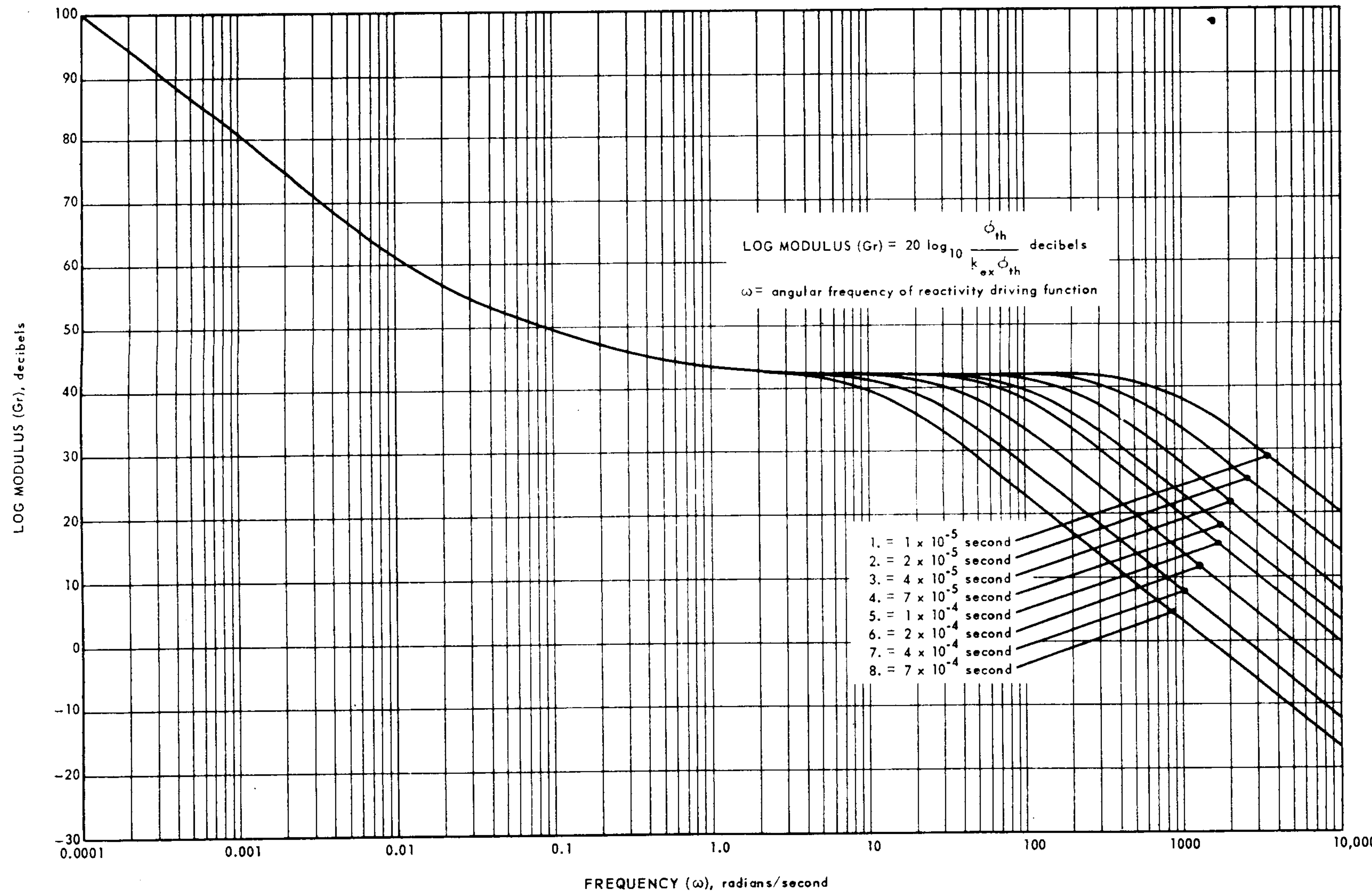


Figure 4.2-11 - Transfer characteristics of nuclear reactors (attenuation)

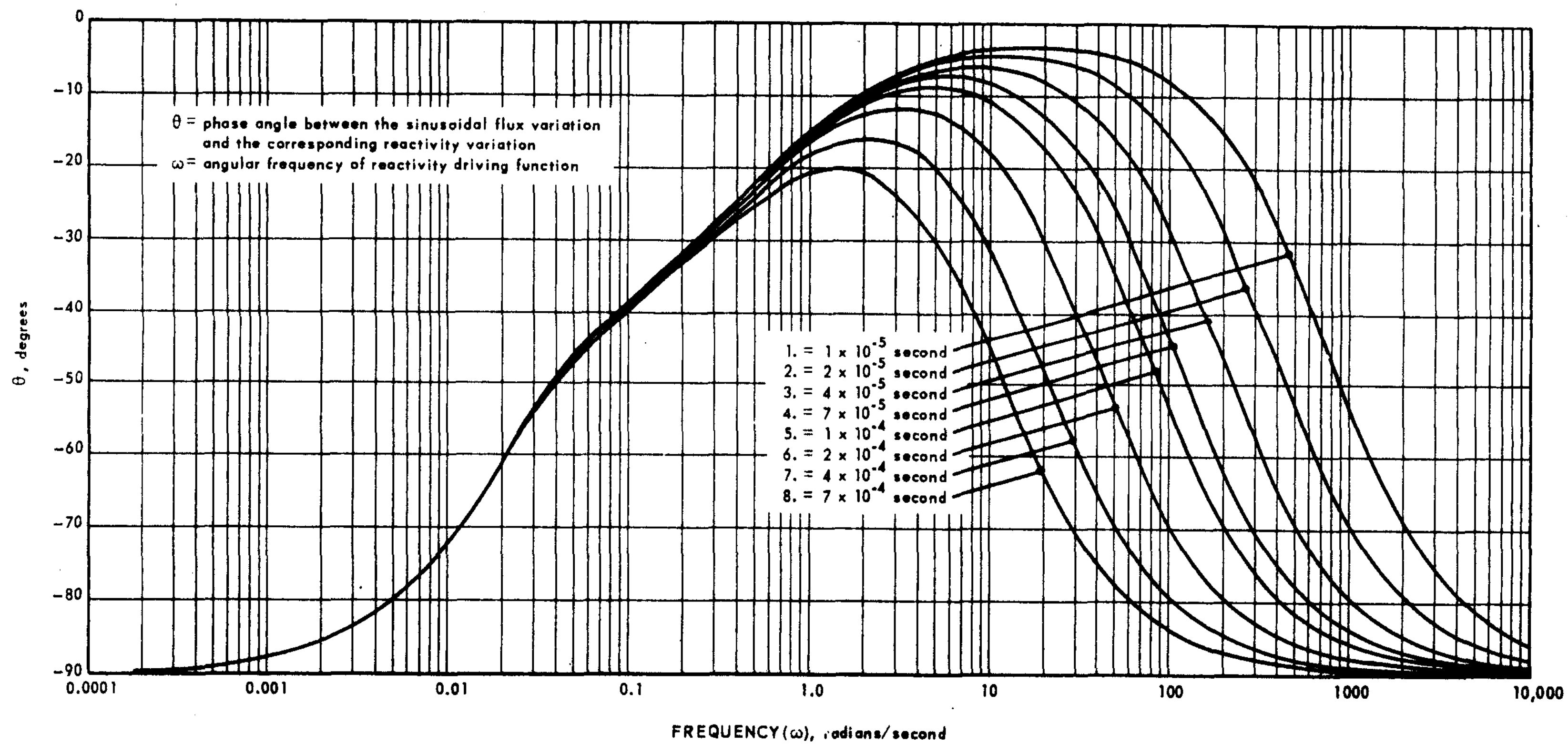


Fig. 4.2-12 – Phase shift transfer characteristics of nuclear reactors

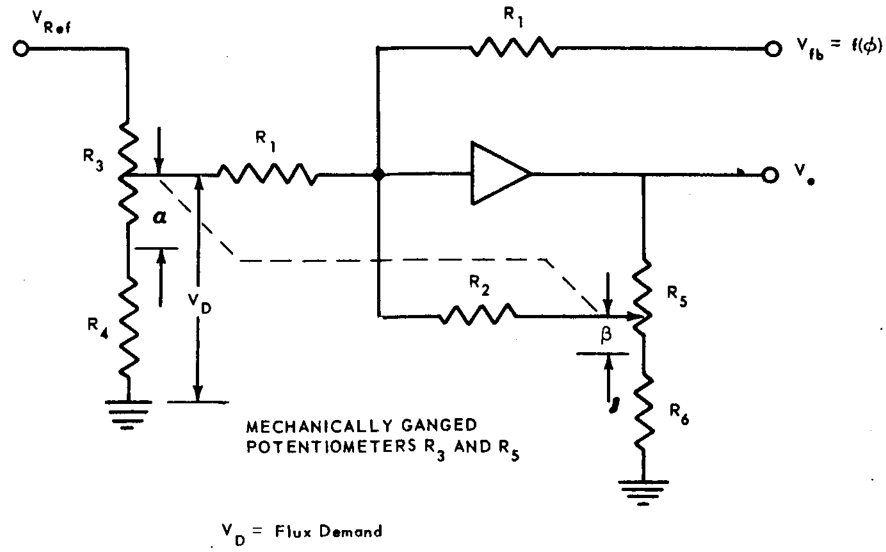


Figure 4.2-13—Flux error amplifier with gain = f (1/ϕ)

Substituting the values

V_e = amplifier Output (O)

$V_D + V_{fb}$ = amplifier Input (I)

$V_D = \phi\%$ during steady state

the equations can now be written.

$$\frac{O}{I} = \frac{R_2}{R_1\left(\dfrac{\beta R_5 + R_6}{R_5 + R_6}\right)}$$

$$\frac{\phi\%}{V_{Ref}} = \left(\frac{\alpha R_3 + R_4}{R_3 + R_4}\right) = \left(\frac{\beta R_5 + R_6}{R_5 + R_6}\right).$$

If $\alpha = \beta$ and $\dfrac{R_3}{R_4} = \dfrac{R_5}{R_6}$

then

$$\frac{O}{I} = \frac{R_2}{R_1}\frac{V_{Ref}}{\phi\%} = \frac{K}{\phi\%} \tag{12}$$

2. **Manual gain variation.** A similar result may be obtained by using a manual switch in place of the ganged gain potentiometer. The reactor operator manually adjusts the gain by positioning a gain switch to correspond to the operating region. This method allows the gain to vary to some degree, depending upon the number of gain steps.

3. **Logarithmic feedback amplifier.**[5] A third method of obtaining a constant gain employs a logarithmic amplifier in the neutron flux feedback so that the output voltage is proportional to the logarithm of the input current:

$$V_0 = K \ln I \tag{13}$$

Where I is the current from the nuclear sensor or ion chamber.

$$I = f\ (\phi\%) \tag{14}$$

The principle of small increments may be used to linearize the equation for the output of the logarithmic amplifier.

The amplifier output voltage is a function of the input current.

$$V = f\ (I)$$

The steady state term plus the incremental gives:

$$V_0 + \Delta V = f\ (I_0) + \left[\frac{df\ (I_0)}{dt}\right]_{I=I_0} \Delta I$$

Substituting in Equation (13)

$$V_0 + \Delta V = K \ln I_0 + \left[\frac{d\ (K \ln I_0)}{dt}\right]_{I=I_0} \Delta I$$

and collecting terms

$$\Delta V = \frac{K}{I_0} \Delta I$$

the amplifier transfer function becomes:

$$G = \frac{\Delta V}{\Delta I} = \frac{K}{I_0} = \frac{K}{\phi\%} \tag{15}$$

Any of these methods of linearization may be employed to obtain an open-loop transfer function with a gain that is independent of reactor power.

With the error amplifier employed as a variable gain integrator, the error amplifier transfer function becomes:

$$G_1 = \frac{K_1}{s\phi\%} \tag{16}$$

where

$$GH(s) = \frac{(0.5)\ (3.5 \times 10^{-4})\ (10) \left(1 + \frac{s}{0.027}\right) \left(1 + \frac{s}{0.4}\right) K_1}{s^2 \left(1 + \frac{s}{0.13}\right) \left(1 + \frac{s}{75}\right) \left(1 + \frac{s}{200}\right) \left(1 + \frac{s}{240}\right)} \tag{17}$$

Let

$$K_1 = \frac{1}{(0.5)\ (3.5 \times 10^{-4})\ (10)}$$

Plots of attenuation and phase margin versus frequency for the open-loop transfer function are shown in Figure 4.2-14.

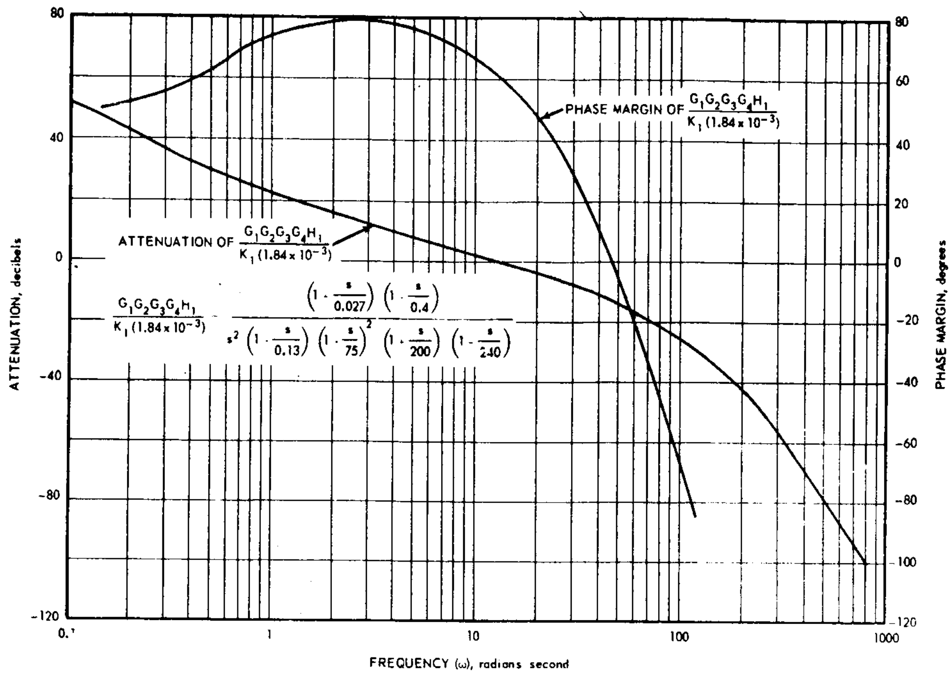


Fig. 4.2-14 – Theoretical open loop frequency response of the flux loop

Heat Transfer

$$G_3 = \frac{\frac{10^3}{W_a}}{1 + \left(\frac{1.2 \times 10^3}{W_a}\right)s} \quad \frac{^oF}{\% \phi} \tag{20}$$

where

W_a = airflow, percent of rated flow. (Rated flow is expressed in pounds per second.)

Thermocouple and Amplifier

$$H_1 = \frac{2.5 \times 10^{-3}}{1 + \frac{2.5s}{\sqrt{W_a}}} \quad \frac{\text{volts}}{^oF} \tag{21}$$

4.3 AUTOMATIC TEMPERATURE CONTROL

In applications of nuclear reactor systems where a reactor is combined with turbomachinery, special problems of temperature control are encountered. A simple flux control is adequate only when the power extraction from the reactor is relatively constant. In this case, the reactor operator may adjust flux level to compensate for long-term variations in power utilization.

It is often necessary, however, especially in cases involving reactors with turbomachinery, to obtain large variations in power while maintaining a relatively constant reactor-coolant temperature. An automatic control that employs reactor outlet temperature as the controlled variable may be used. This temperature control employs the closed flux loop (8.5.3) as a forward element. The demanded temperature is compared to the reactor outlet temperature and the difference used to manipulate the reactor flux to force the measured and demanded temperature to correspond. Since reactor outlet temperature is an indication of the energy being delivered by the reactor, assuming that gas flow is known, reactor discharge temperature is often used as the controlled variable. A temperature-error integrator may be employed to improve the accuracy of the temperature loop. There is a time lag associated with the heat transfer from the fuel elements to the coolant and from the coolant to the thermocouple that may be cancelled by adding a lead function to the error amplifier.

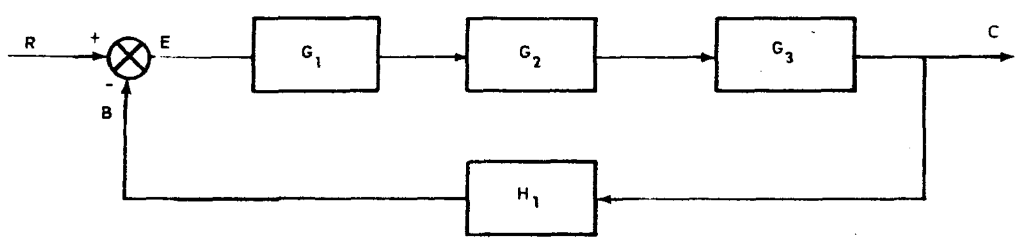
A temperature control loop is shown in Fig. 4.3-15. Typical transfer functions as shown below are assumed:

Error Amplifier

$$G_1 = \frac{8\left(1 + \frac{s}{0.083}\right)}{s} \ \frac{\text{volt}}{\text{volt}} \tag{18}$$

Closed Flux Loop

$$G_2 = \frac{1}{\left(1 + \frac{s}{20}\right)\left(1 + \frac{s}{75}\right)^2\left(1 + \frac{s}{200}\right)\left(1 + \frac{s}{240}\right)} \ \frac{\%\,\phi}{\text{volt}} \tag{19}$$



R = reference input, volts, dc
B = feedback signal, volts, dc
E = loop error, volts, dc
C = control variable (outlet temperature)

G_1 = transfer function of stabilizing elements, volts/volt
G_2 = transfer function of flux loop, % ϕ/volt
G_3 = transfer function of reactor heat transfer, $\Delta T/\Delta\,\%\phi$
H_1 = transfer function of sensor and amplifier, Δ volts/ΔT

Fig. 4.3-15 – Block diagram of temperature control loop

The closed-flux loop-transfer function is obtained by the approximate method:

$$\frac{O}{I} = \frac{1}{H} \text{ when GH} > 1$$

$$\frac{O}{I} = G \text{ when GH} < 1$$

The open-loop temperature transfer becomes a function of the airflow through the reactor. The open-loop transfer function may be evaluated for various values of airflow. Assuming a normal operating region of 40 percent to 100 percent airflow, the transfer functions are indicated for these two extremes. Higher frequency terms have been neglected since they have very little effect over the frequency range of interest.

When airflow is 40 percent:

$$GH \approx \frac{0.5\left(1 + \frac{s}{0.083}\right)}{s\left(1 + \frac{s}{0.033}\right)\left(1 + \frac{s}{2.53}\right)\left(1 + \frac{s}{20}\right)\left(1 + \frac{s}{75}\right)^2} \qquad (22)$$

When airflow is 100 percent:

$$GH \approx \frac{0.2}{s\left(1 + \frac{s}{4}\right)\left(1 + \frac{s}{20}\right)\left(1 + \frac{s}{75}\right)^2} \qquad (23)$$

The attenuation frequency plot of this function, shown in Fig. 4.3-16, indicates a crossover frequency of 0.18 radians per second and a phase margin varying from 70 degrees for W_a = 40 percent flow to 87 degrees for W_a = 100 percent rated airflow.

Fig. 4.3-17 shows the temperature loop transient response of the system when operating initially at 63.3 percent FP. A stable system is indicated with a time constant of

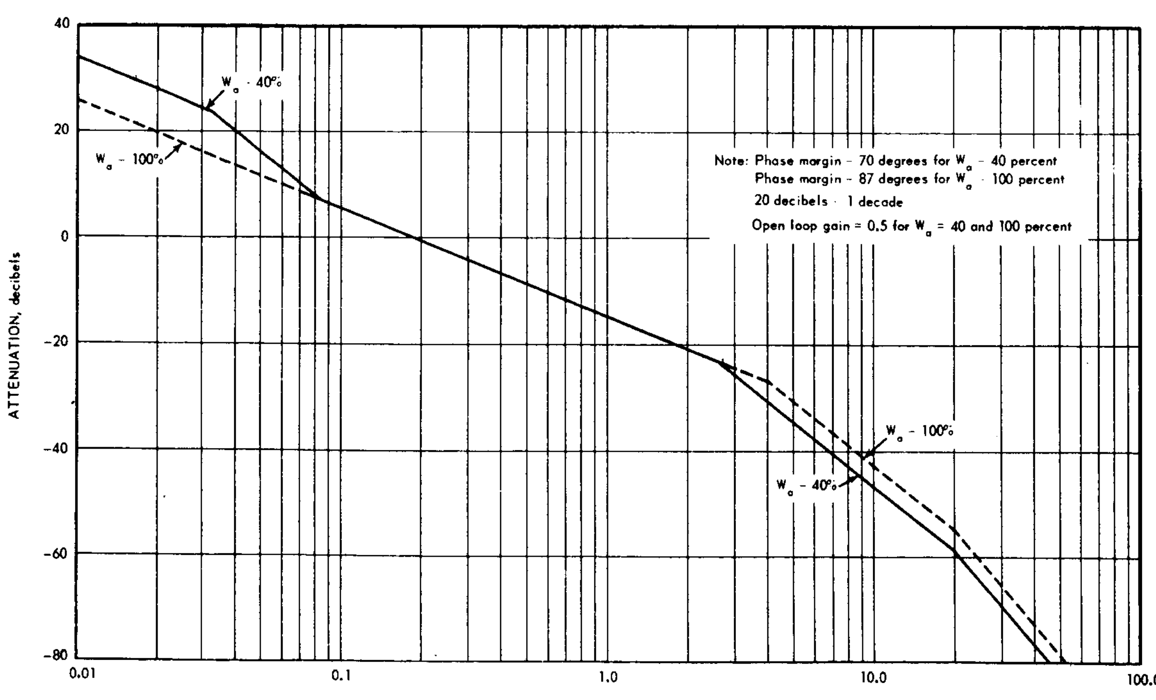


Fig. 4.3-16 – Attenuation frequency characteristics of the open loop transfer function of the temperature loop

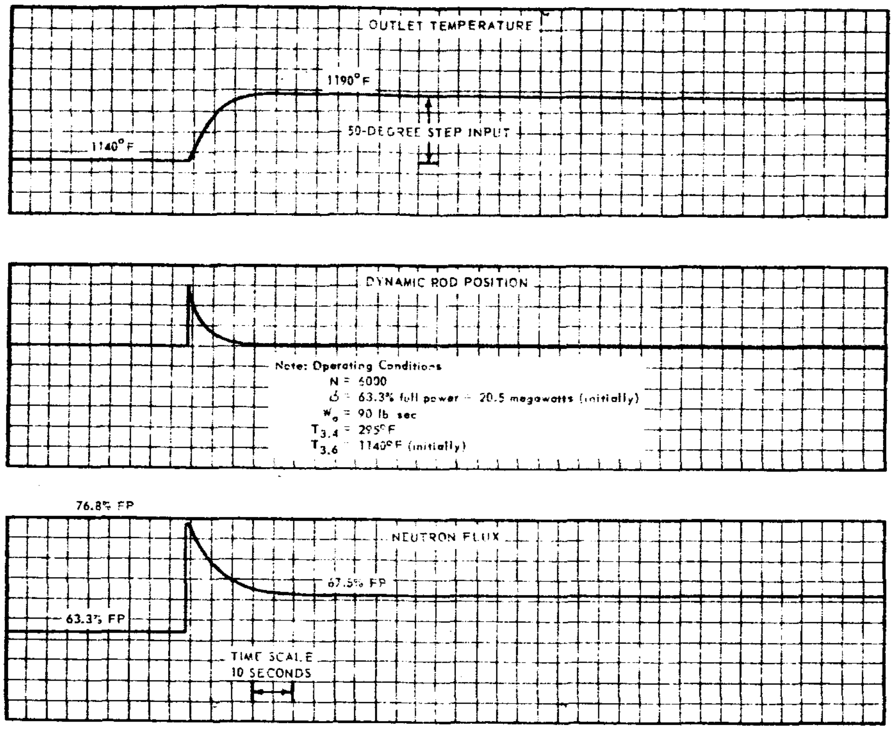


Fig. 4. 3–17 – Temperature loop transient response of the system when operating initially at 63.3 percent full power

approximately five seconds. The integrating action of the temperature-error amplifier maintains essentially zero steady state error. This temperature control system is adequate for a test reactor operated over a moderate range of compressor inlet temperatures and temperature conditions as found during ground tests and limited flight tests.

The design of this system cannot be considered an optimum design for a temperature loop because it requires an input signal from the flux demand and because, unless logarithmic feedback gain compensation is used, limitations are placed upon the operator or application of the system. The mechanization scheme, as described, is worthwhile when equal use is made of flux and temperature control. However, if the operation of the reactor is to be conducted primarily on temperature control, a different temperature loop should be used. Such a system includes continuous gain compensation and permits the temperature error to maintain complete control of the flux loop.

4.4 STARTUP CONTROL

The level below the power range is referred to as startup range and can span from reactor shutdown power, source level of 10^{-10} FP, to the lower extreme of the power range. Some overlapping of the two systems occurs, and in some cases, the startup system completely overlaps the power range. Although automatic startup systems have been developed, an automatic system is not normally required. An automatic startup system is useful in the situation where the number of operators is limited, where time is an important factor in the operation, and where the operator is required to fulfill other responsibilities as the reactor is started up. In most cases, however, time is not critical and an operator has only the responsibility of bringing the reactor to power. Thus, a manual system is sufficient. There is very little difference between the mechanization of a manual and an automatic startup system. A complete automatic startup system can be implemented by utilizing the power-range flux loop and by adding a demand device to the manual startup circuitry.

The startup range is sometimes divided into two operating ranges: the source range and the intermediate range. The source range operation is in the region of source level to 3-to-4 decades above source level. The intermediate range covers the region between the source range and the power range, spanning about 5-to-6 decades of reactor operation. The ability to indicate power over such a broad spectrum of reactor power is usually accomplished by operating on the logarithm of reactor power. This contracts the amplitude of the measured signal to the indication system but does so at the loss of accuracy.

In the power regions near source level, power is measured by a fission chamber or proportional counter, which counts the individual fissions occurring within the chamber itself. Thus, if the chamber is immersed in a representative nuclear field, it counts the number of fissions occurring within the chamber to produce an output proportional to reactor power. The signal is in the form of discrete pulses with a pulse rate proportional to the reactor power. Assuming that the pulse rate is 1 pulse per second at source level, and that the power range begins 8 decades above the source level, the count rate is 10^8 counts per second at the lower portion of the power range. Present day electronic components are incapable of operating at these pulse rates. Although transistorized circuits operating to 2×10^6 counts per second have been designed and used, another factor compounds the situation. Because the fissions occurring within the fission chamber are random in period, the pulse rate fluctuates wildly below approximately 20 counts per second. This fluctuation is referred to as statistical noise. At count rates above 20 counts per second, statistical smoothing is obtained. This requirement for at least 20 counts at source level indicates a count rate of 2×10^9 at the power range. This high-pulse-rate difficulty may be overcome by cascading two or three pulse systems or count rate systems. The nuclear sensors are then staggered so that their sensitivities differ by 3-to-4 decades. Thus, as one sensor channel is about to saturate, a second level of instrumentation begins to operate. Since the low level pulses, about 100 microvolts for a fission chamber and about 1 millivolt for a proportional counter, are in the induced noise region, the transmission of the signal from the nuclear sensor to the electronic equipment presents another problem. The lack of sufficient reactor shielding may require remote stationing of the operator and the electronic components. As much as 200 feet may be required between the reactor and the operator and components. The design and installation of a good low noise cable system between the reactor and the control equipment becomes necessary. The use of pulses has other disadvantages. A considerable amount of equipment is required to measure reactor power with log-count-rate techniques, as shown in Fig. 4.4-18a. Transmitting the low level signals over long cables may require a preamplifier near the chamber to increase the signal-to-noise ratio. A pulse amplifier, a discriminator to remove gamma pulses, a pulse shaper to standardize the pulse shape and size, a device to furnish a signal proportional to the logarithm of the pulse rate are required. A period computer and amplifier are also usually included in the count rate equipment.

Because of the pulse problem, some source-range systems have been designed to cover only 3 or 4 decades or one level of instrumentation requiring the addition of an intermediate range system or log flux system. The intermediate range system employs a nuclear sensor which provides an analog output signal proportional to nuclear power. These sensors are normally gamma-compensated so that they measure only neutron flux. This type of system, requiring only a log amplifier, is considerably simpler than the count-rate system. A log-flux system, shown in Fig.4.4-18b, does not have the inherent noise problem of the pulse system because filtering can be added to the analog signal without destroying the signal. The cabling system problem is still critical due to the low-level d-c signals of approximately 10^{-11} to 10^{-3} amperes.

While the log flux system offers the advantage of mechanization, simplicity and less noise susceptibility, it has the disadvantage of being more affected by gamma flux. This gamma sensitivity is a major obstacle when the gamma flux is high compared to the neutrons, as is the case with some shutdown reactors. The log-count-rate system, because of the discriminator, can be designed to be gamma-insensitive even for the high gamma-to-neutron ratios for the shutdown reactor. It is difficult to design gamma-compensated ion chambers for the high gamma-neutron ratios. Therefore, the possible application of a log-flux system in the source range is predicated upon the reactor. If, during the shutdown condition, the gamma-to-neutron ratio is low, as when a beryllium reflector is used, the log flux system should be used and will provide much less difficulty than the pulse system. Two staggered systems may be required to span the entire startup range, but the advantages still outweigh the problems of the log-count rate system. If, however, the relation of gammas to the neutron flux levels is high, use of the log-flux system may be impossible. Use of the system should not be discarded, however, without prior investigation.

A third method exists for reactor startup. In this system, shown in Fig.4.4-18c, a sensitive amplifier is placed in the feed-back loop to amplify the ion chamber signal. This micromicroammeter usually has a decade gain switch to enable the amplifier gain to be changed by factors of ten, although some amplifiers have intermediate gain steps. This process is similar to the second step for maintaining the flux loop gain constant described in section 4.2 . The micromicroammeter allows the power range flux control to be operated in the very low power regions. Although the operator must change the gain for each decade of power change, this is not a major disadvantage, especially for the systems where relatively few reactor startups are made. The use of the micromicroammeter does not normally replace all of the source-range equipment, however, because the amplifier cannot operate at very low signal levels which occur in the source range. Thus, a low-level system is necessary for the initial reactor startup when the reactor is cold-clean. This limitation is a function of the extremely high gain required and the resultant susceptibility of the amplifier to background electrical noise. The minimum signal levels to which the system is to be exposed determines the quality of the signal transmission system required.

The micromicroammeter may also be used to check out the power range flux control system, as shown in Fig.4.4-18c. The automatic flux control system can be checked for stability and performance with the reactor operating at low power levels where damage cannot result unless control of the reactor is lost, resulting in a power rise of several decades.

4.5 REACTOR SAFETY

The application of nuclear power has resulted in increased emphasis on automatic protective systems. The reactor control system, because of the excess reactivity required for xenon override, fuel depletion and temperature coefficients, has the inherent capability of introducing reactivity many times that required to render the reactor prompt critical.

The design of a safety system must be based upon the consideration of all possible component and operator malfunctions and errors, including those situations that appear to be quite improbable. Perturbations from external causes, such as loss of power in the operating facility, and from natural causes, such as earthquakes should also be considered.

Some levels of safety employ all or part of the automatic servosystem, while others bypass the servoloop and operate directly on the control actuators. Some safety designs employ separate safety actuators. In other cases, safety methods completely independent of the normal control methods of inserting negative reactivity are used.

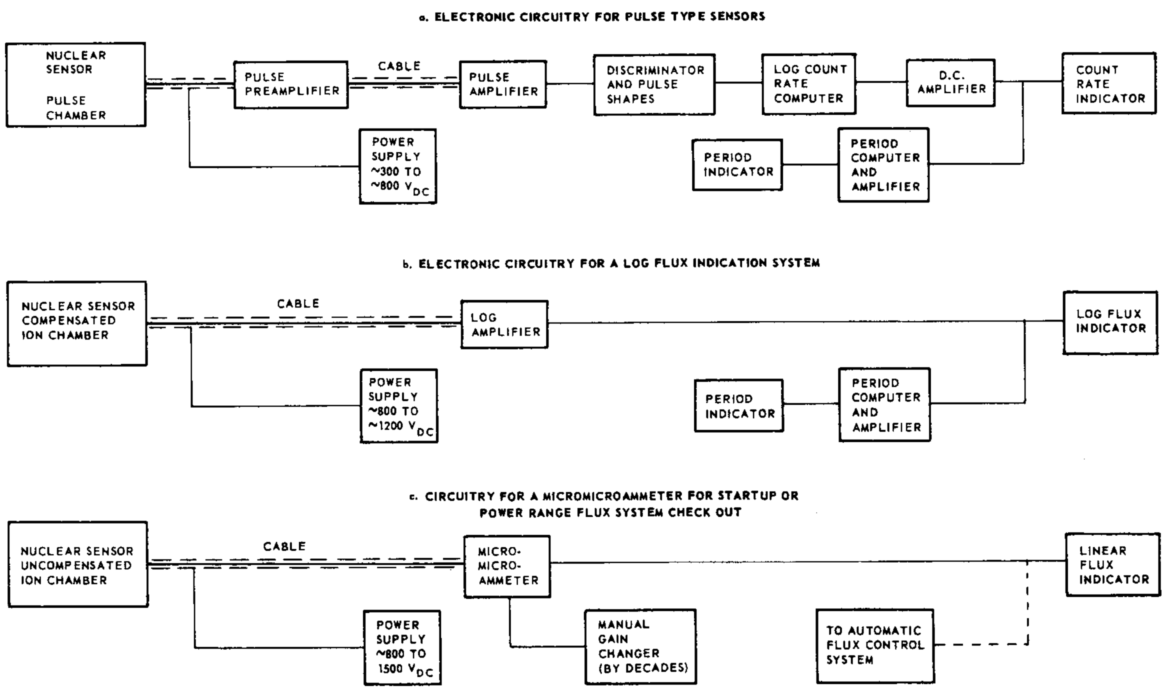


Fig. 4.4-18 – Possible circuits for startup indication

Since a complex safety system may include several levels of safety and since each safety action may be initiated by a large number of reactor system parameters, the safety system normally includes an indicating panel, (annunciator), to inform the operator of the type of protective action imposed and to indicate which of the parameters initiated the safety function. The annunciator panel may retain a record of the offending parameters until the operator acknowledges the situation and acts to clear the memory device.

Some of the safety actions are designed to prevent recalling until the cycle is completed. A complete reactor shutdown results. An example of this action is reactor scram. In other levels of automatic safety, the imposed response may be removed automatically when all parameters are returned to their normal operating ranges. In still other safety actions, the imposed response is removed by operator acknowledgement and some positive action.

4.5.1 Parameter Limiting

Parameter limiting, shown in Fig.4.5-19, encloses one of the automatic servoloops with another external loop. This loop functions as any other external loop except that it is restricted to only one demand polarity to the internal loop. Normally, the reference to the limiter loop remains fixed, at least for a given set of operating conditions. As the measured parameter begins to differ from the reference in a direction determined to be unsafe, the amplified difference of the proper polarity is added to the automatic loop to return the limiting parameter to a safe operating level. As long as the limiting parameter is within the safe operating region, the amplified error signal is blocked. The limiter loop is essentially an open loop as long as the limiter level is within the predetermined safe region of operation. Because limiter action results in little or no noticeable perturbation in the power plant operation, it offers the optimum type of safety action. Limiter action relies, however, upon the ability of the automatic system to be operational. Hence, this safety response should not be applied to those conditions which can result from control system malfunctioning. This safety action is normally employed when two parameters are related by a variable and one parameter may exceed its pre-set limit, although the other parameter is controlled within the tolerable limits. An example of this condition is the relationship between fuel element temperature and reactor discharge temperature where $T_{fe} \cong T_g + f(W_g^{0.2})$. An increase in gas flow results in an increase in reactor power and fuel element temperature, where reactor discharge temperature, T_g, is the controlled parameter and W_g is gas flow.

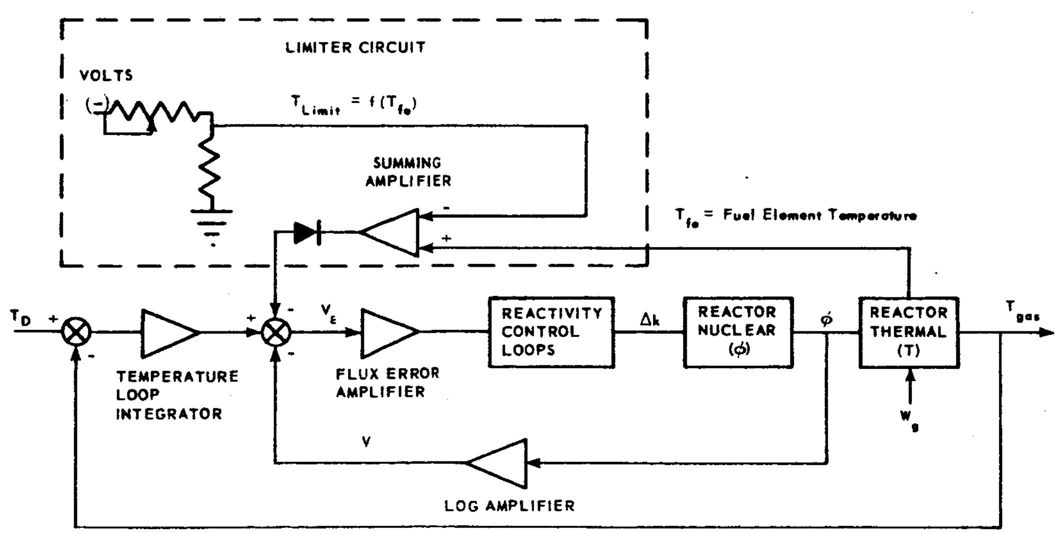


Fig. 4.5-19 – Flux loop with fuel element temperature limiting

4.5.2 Alarm

Alarm does not directly affect the power plant operation, serving only to alert the operator.

The signal being monitored is compared to a pre-established reference and, as the monitored signal differs from the reference in the direction toward unsafe operation, the operator is given a visual or audible indication. The operator, aware that an abnormal condition exists, monitors the related parameters and attempts to ascertain the cause and possible results.

4.5.3 Interlock

Interlocks are designed to prevent unsafe situations that may arise because of operator or control system error. Interlock is normally used to insure the following of a defined sequence of operation by the operator or automatic control system. Interlocks may be designed to prevent continuation of the operating sequence, or to cancel a command which might influence the operation in an unsafe direction. For example, startup interlocks prevent the insertion of positive reactivity until all auxiliaries and power supplies are turned on. An interlock protects an auxiliary pump by preventing the energizing of the pump drive unless certain valves are in the proper position for safe pump operation. An interlock may also serve to remove certain safety responses in a given operating region.

4.5.4 Override

Override safety is an active safety which removes power of control from the servosystem or the operator, and places its own command for a defined change in operating level. Override may take several forms in its application:

1. Setback - A fixed magnitude of bias is added to the control system influencing the system toward safe operation. Setback may also change the operating point to some pre-selected value determined to be safe for forseeable situations.
2. Shutdown - Reactivity is removed to place the reactor on a negative period of decreasing power. The reactor will reach some steady-state period and continue to decrease until another action is taken by the operator or the safety system to overcome the decrease in power and to restore, perhaps at a new operating level, normal nuclear system operation.

Both override actions should be applied to the safety system design with caution. In a power system that also furnishes the energy to drive its own coolant circulating pumps, a small or slow decrease in power under certain conditions may not be a protective measure. If coolant-blower speed is a function of reactor power, some operating point exists where a decrease in reactor power results in instability or where blower speed and coolant flow will decrease faster than the power. Such a response would result in fuel element overtemperature. Normally the blower airflow is relatively independent of reactor power over a broad range of the operating region and will not result in the situation described. The possibility of such an overtemperature, however, should be investigated.

4.5.5 Scram

Scram is an override function of the greatest severity. Negative reactivity is forced into the reactor at a maximum rate, sufficient to reduce reactor power effectively to zero. Once a scram function is initiated it cannot be recalled until the cycle has been completed. Scram is normally employed as the final level of power plant protection. This function is mechanized with a minimum number of components. It bypasses as much of the control system as possible to insure maximum reliability.

A list of power plant parameters is presented in Table 1. Possible safety actions and levels are also indicated. Each power plant parameter must be considered individually both to determine the consequence of its falling outside pre-set limits and its value as an anticipatory indication of impending power plant damage. The effects of certain malfunctions, the methods of anticipating such conditions, and the means for preventing or restricting the consequences can be analyzed by logic, mathematical analysis, or by simulation.

TABLE -1

SAFETY RESPONSES

Parameter		Scram Trip		Alarm Trip
1. Fuel element temperature	(3)[a]	Normal - 100°F, adjustable	(3)	Normal - 50°F
2. Air temperature, turbine exit	(3)	Normal - 100°F, adjustable	(3)	Normal + 50°F
3. Power level, high-power	(2)	10 to 125%, adjustable		-
4. Power level, low-power	(2)	1 to 4%, if armed		-
5. Switching monitor	(10)	Any channel open		-
6. Period, startup range only	(3)	5 second, and 2 out of 3 coincidence, if armed	(3)	10 second, any channel, if armed
7. Engine speed	(1)	105% N		-
8. Temperature mode, low-power		Both low-power trips unoperated		-
9. Facility monitor		Normal operating background + 200%		Normal operating background - 100%
10. Low actuator pressure		70 to 90%		-
11. Power demand		Positioned above 1% with 1% transfer relay unoperated, if armed		-
12. 400 cycle voltage		±10%		±5%
13. 400 cycle frequency		±10%		-
14. 28 V, d-c bus		-6 V		-
15. 1500 V, d-c bus		-300 V		-
16. ±200 V, d-c bus, startup only		20 V decrease, if armed		-
17. 280 V, d-c bus, startup only		-30 V, if armed		-
18. 60 cycle voltage		-10%, micromicroammeter operation only		-
19. Operator discretion		At will		-
20. Emergency engine shutdown		At will		-
21. Test sets		Connector uncovered		-
22. Key switch		Unoperated		-
23. Shim grate		-	(8)	Any unlatched
24. Equipment continuity		-		Any interlock open
25. Ion chamber cable continuity		Cable improperly connected		-

[a]Numbers in parenthesis indicate redundant trips.

5.0 Reactor Nuclear Characteristics Simulation

The dynamic analysis of the reactor control system and the overall power plant is best accomplished using general purpose electronic analog computers. A large amount of equipment, approximately 150 amplifiers and associated nonlinear equipment, may be required for a complete simulation of a reactor and its associated propulsion system. The detailed simulation provides an excellent means of designing an optimum control system; permits operator training; provides the capability of analyzing the effects of component failures,[6] and is a very good method of testing control hardware prior to field operation. The basic simulations required include the reactor kinetics, reactor heat transfer characteristics, the propulsion system that acts as an air pump, and the control system.

There are several methods of simulating reactor kinetic equations but the most satisfactory, from the standpoint of control system development, is the transformation of the kinetic equations into a simple transfer function.[7] The kinetic equations for a thermal reactor[8] are as follows:

$$\frac{d\,\phi_{th}}{dt} = \phi_{th}\left(\frac{k_{ex} - \beta}{l^*}\right) + \frac{P_1 \sum_{i=1}^{6} \lambda_i\, r_i\,(t)}{l^* \Sigma_{Ath}\,(1 + L^2 B^2)} \tag{24}$$

$$\frac{dr_i\,(t)}{dt} = \frac{\beta_i\ \gamma \Sigma_{Fth}}{1 - \alpha_1}\ \phi_{th} - \lambda_i r_i\,(t) \tag{25}$$

Symbols used to indicate reactor characteristics are shown in Table 5-1.

TABLE 5-1

DEFINITION OF SYMBOLS USED FOR REACTOR CHARACTERISTICS

ϕ_{th} =	effective thermal neutron flux as a function of time
i =	one of the six delayed neutron groups
$\beta = \sum_{i=1}^{6} \beta_i$ =	effective fraction of delay neutrons contributing to the chain reaction as compared to the total number of effective neutrons
l^* =	neutron generation time
$1 - \alpha_1$ =	the fraction of fissions due to thermal neutrons for a reactor that is near critical
Σ_{Ath} =	thermal absorption cross section
Σ_{Fth} =	thermal fission cross section
B^2 =	geometrical buckling constant
L =	thermal diffusion length
p =	operator = d/dt or $J\omega$ depending on driving function
P_1 =	average probability that a neutron starting at fission spectrum energies will slow down to thermal energies
λ_i =	delayed neutron decay constant of the i^{th} group
γ =	number of neutrons per fission
r_i =	concentration of precursors of the i^{th} group
k_{ex} =	excess reactivity

To transform these equations into transfer function form[7] let:

$$k_1 = \frac{\gamma \Sigma_{Fth}}{1 - \alpha_1} \tag{26}$$

$$k_2 = \frac{\Sigma_{Ath}\,(1 + L^2 B^2)}{P_1} \tag{27}$$

Steady state operation requires $k_{ex} = 0$, $\frac{d\phi_{th}}{dt} = 0$, $\frac{dr_i}{dt} = 0$.

Equation (24) and (25) at steady state indicate the conditions required for a reactor which is just critical. This relation is shown:

$$\sum_{i=1}^{6} \frac{\lambda_i r_i}{k_2} = \beta\phi_{th} \tag{28}$$

$$\lambda_i r_i = \beta_i k_1 \phi_{th} \tag{29}$$

Substitution of (29) into (28) results in

$$\frac{k_1 \phi_{th} \sum_{i=1}^{6} \beta_i}{k_2} = \beta\phi_{th} = \sum_{i=1}^{6} \beta_i \phi_{th} \tag{30}$$

Therefore k_1 must equal k_2.

When Equations (24) and (25) are re-written in operator form with the substitutions of k_1 and k_2, from Equations (26) and (27).

$$p\phi_{th} = \left(\frac{k_{ex} - \beta}{l^*}\right)\phi_{th} + \frac{1}{k_2 l^*} \sum_{i=1}^{6} \lambda_i r_i \tag{31}$$

$$pr_i = \beta_i k_1 \phi_{th} - \lambda_i r_i \tag{32}$$

or

$$r_i = \frac{\beta_i k_1 \phi_{th}}{p + \lambda_i}$$

Substitution of (32) into (31) gives:

$$p\phi_{th} = \frac{k_{ex}\phi_{th}}{l^*} - \frac{\beta\phi_{th}}{l^*} + \frac{k_1}{k_2 l^*} \phi_{th} \sum_{i=1}^{6} \frac{\lambda_i \beta_i}{p + \lambda_i} \tag{33}$$

$$p\phi_{th} = \frac{k_{ex}\phi_{th}}{l^*} + \frac{\phi_{th}}{l^*} \sum_{i=1}^{6} \frac{\lambda_i \beta_i - \beta_i (p + \lambda_i)}{p + \lambda_i} \tag{34}$$

$$p\phi_{th} = \frac{k_{ex}\phi_{th}}{l^*} - \frac{p\phi_{th}}{l^*} \sum_{i=1}^{6} \frac{\beta_i}{p + \lambda_i} \tag{35}$$

The input to a reactor is a change in reactivity = k_{ex}, and the output is the effective neutron flux level, but the effectiveness of k_{ex} depends on power level. Hence, the transfer function for the reactor must be expressed as $\Delta\phi_{th}/\phi_{th}/k_{ex}$.

When this expression is rearranged,

$$\frac{\frac{\Delta\phi_{th}}{\phi_{th}}}{k_{ex}} = \frac{1}{p\left[l^* + \sum_{i=1}^{6} \frac{\beta_i}{p + \lambda_i}\right]} = KG_R(p) \tag{36}$$

From the above equations, it may be concluded that for steady state, when $k_{ex} = 0$, the equation for criticality holds; i.e., $\frac{\nu\Sigma_{Fth}}{1 - \alpha_1} = \frac{\Sigma_{Ath}(1 + L^2 B^2)}{P_1}$ or $k_1 = k_2$, which is an equation of condition in terms of nuclear characteristics and geometry of the reactor, and has no effect on the kinetics of the reactor. The time behavior of the reactor is primarily dependent upon the concentration and mean lifetime of the delay neutrons, provided $k_{ex} < \beta$, although the generation time influences prompt changes that occur during step inputs.

The reactor kinetic transfer function can be simulated by means of a d-c amplifier with the proper input and feedback impedances.

Fig. 5-19 shows a stabilized high gain d-c amplifier with parallel impedance elements in the feedback in which:

$$Z_f = \frac{1}{\frac{1}{Z_0} + \frac{1}{Z_1} + \frac{1}{Z_2} \cdots \frac{1}{Z_n}} = \frac{1}{\frac{1}{Z_0} + \sum_{k=1}^{n} \frac{1}{Z_k}} \tag{37}$$

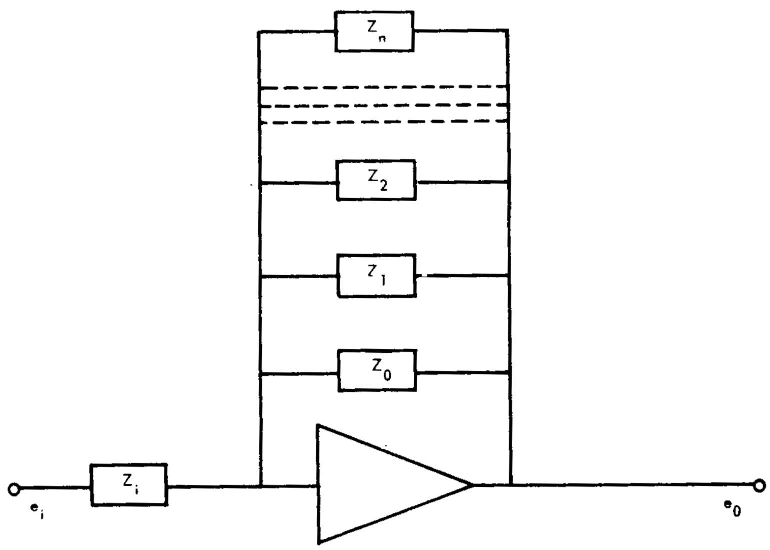


Fig. 5.19 - High gain amplifier with parallel impedances in the feedback circuit

$$\frac{e_0}{e_i} = -\frac{Z_f}{Z_i} = -\frac{1}{Z_i\left[\frac{1}{Z_0} + \frac{1}{Z_1} + \frac{1}{Z_2} \cdots \frac{1}{Z_k}\right]} = -\frac{1}{Z_i\left[\frac{1}{Z_0} + \sum_{k=1}^{n} \frac{1}{Z_k}\right]} \tag{38}$$

Where Z_0 is a capacitor ($Z_0 = 1/pC_0$); and Z_k is a resistor and a capacitor in series ($Z_k = R_k + 1/pC_k$); and Z_i is a resistor R_0 then Equation (38) takes the form of Equation (39).

$$\frac{e_0}{e_i} = -\frac{Z_f}{Z_i} = -\frac{1}{R_0\left[pC_0 + \sum_{k=1}^{n} \frac{1}{R_k + \frac{1}{pC_k}}\right]} \tag{39}$$

$$\frac{e_0}{e_i} = -\frac{1}{p\left[R_0 C_0 + \sum_{k=1}^{n} \frac{\frac{R_0}{R_k}}{p + \frac{1}{R_k C_k}}\right]} \tag{40}$$

The reactor transfer function about an operating level ϕ Equation (36), is identical in form to Equation (40).

By letting

$$\lambda_i = \frac{1}{R_k C_k}, \quad \beta_i = \frac{R_0}{R_k} \text{ and } l^* = R_0 C_0$$

an accurate analog of a reactor transfer function about a given operating level is obtained. By employing a multiplier, as shown in Figure 5-20 it is possible to remove the last restriction so that a circuit similar to that of Figure 5-20 also represents the reactor throughout the power range for large perturbations in flux level. The Zener diode is used in the feedback path to assure that the reactor is always started up with the proper polarity and to protect the amplifier from overloading.

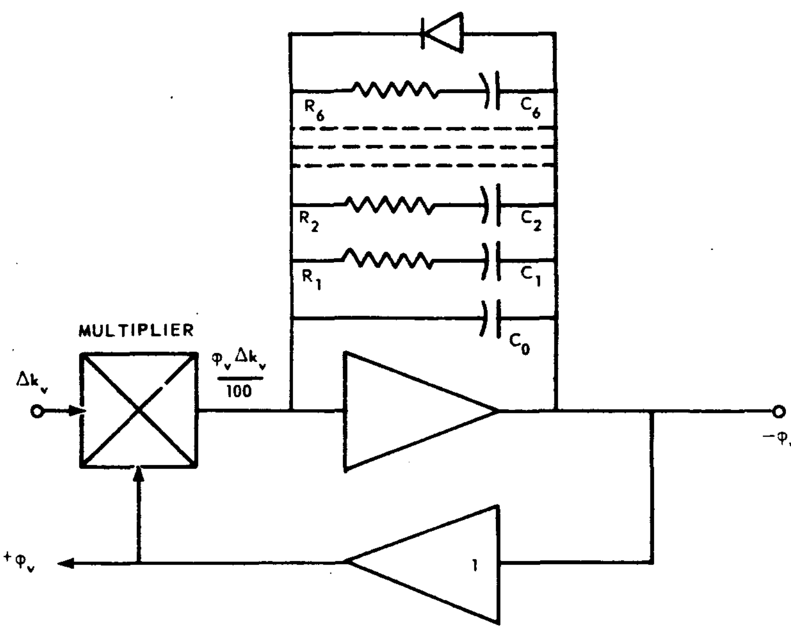


Fig. 5.20 - Reactor kinetic simulator with no temperature coefficient of reactivity

The ladder network, shown in Figure 5-20, may be constructed as a compact package with plug-in connections. This package, which may be used for several simulator setups, prevents the resistors and capacitors from cluttering the computer problem board.

The decay constants λ_i and the delayed fractions β_i (actual) for the six delayed groups are fixed by the fuel used in the chain reaction. The neutron generation time l^* is determined by the type of reactor, while the delayed fraction β_i (effective) is also influenced by the reactor design. Table 5-2 indicates the decay constants and the effective delayed fractions for an epithermal reactor.[9]

To permit the addition of actual control components to the reactor simulator, the simulation must be based upon real time. The portion shown in Figure 5-20 should be an actual simulation of the neutron kinetics on a one-to-one time scale.

The scale factors influence the size of resistors and capacitors that are needed in the reactor-kinetics simulator. The simulator transfer function must be approximately 1 percent that of the reactor if reasonable values of resistance and capacitors are to be used. The input $\Delta k/k$ must be raised 100 times over normal. Voltage of variables should be restricted, if possible, to values between 5 and 100 volts. Because there is some choice in

TABLE 5-2

DELAYED NEUTRON CONSTANTS
USED FOR REACTOR SIMULATOR

Delayed Group	β_i (effective)	λ_i
1	0.00014	5.332
2	0.00105	1.507
3	0.00316	0.3180
4	0.00149	0.1181
5	0.00162	0.0319
6	0.00028	0.0128
$l^* = 2.0 \times 10^{-5}$ seconds.		

setting scale factors, scale factors that relate the machine variables to the problem variables by round numbers should be assigned to simplify conversion. Let subscript v represent the voltage or machine variables; the problem variables have no subscript.

A scale factor of 10,000:1 $= (\Delta k/k)_v$ divided by $\Delta k/k$, is assigned so that 100 volts $(\Delta k/k)_v$ = 1 percent actual reactivity. Since the multiplier has a built-in scale factor of 0.01, it is necessary to set the simulator reactor transfer function $KG_{RS} = (1/100)KG_R(p)$. Reasonable values of resistors and capacitors can then be used with available amplifiers.

In Figure 5-2, let:

$$\frac{E_0}{E_i} = \frac{KG_R(p)}{100} = \frac{1}{100p\left[R_0C_0 + \sum_{k=1}^{6} \frac{\frac{R_0}{R_k}}{p + \frac{1}{R_kC_k}}\right]} \tag{41}$$

The calculations for the circuit parameters are as follows:[10]

$$R_kC_k = \frac{1}{\lambda_k}; \; R_1C_1 = \frac{1}{\lambda_1} = \frac{1}{5.332} = 0.1875 \text{ seconds}$$

$$\frac{R_0}{R_k} = K_{sf}\beta_k; \; \frac{R_0}{R_1} = K_{sf}\beta_1 = 100(0.00014) = 0.014$$

Where K_{sf} is the scale factor, $K_{sf} = 100$.

In order that R_1 through R_6 will not have excessively large values, and C_1 through C_6 too small values, let R_0 be greater than 0.2 but less than 0.5 megohms:

$$R_0 = \frac{K_{sf}\beta_k}{\lambda_k C_k} = \frac{K_{sf}\,\beta_1 \frac{1}{\lambda_1}}{C_1} = \frac{K_{sf}\,\beta_1}{\lambda_1 C_1} = \frac{0.014 \times 0.1875}{0.0082} = 0.32012$$

Let

$$R_0 = 0.32 \text{ megohms}$$

$$R_k = \frac{R_0}{K_{sf}\beta_k}$$

$$R_1 = \frac{R_0}{K_{sf}\beta_1} = \frac{0.32}{0.014} = 22.86 \text{ megohms}$$

Let

$$R_1 = 22.9 \text{ megohms}$$

$$C_1 = 0.0082 \text{ microfarads}$$

These numbers can now be checked:

$$R_k C_k = \frac{1}{\lambda_k}; \; R_1 C_1 = \frac{1}{\lambda_1}$$

$$(22.9)(0.0082) \cong \frac{1}{5.332}$$

$$0.18778 \cong 0.18755$$

Table 5-3 contains circuit values for the reactor simulator.

TABLE 5-3

CIRCUIT VALUES FOR REACTOR SIMULATOR

K	R_kC_k, seconds	R_0/R_k	C_k, microfarads	R_0, megohms	R_k, megohms
1	0.1875	0.014	0.0082	0.320	22.9
2	0.6635	0.105	0.22	0.3195	3.0
3	3.1446	0.316	3.1	0.3205	1.01
4	8.4674	0.149	4.0	0.3202	2.12
5	31.3479	0.162	15.8	0.3204	1.98
6	78.125	0.028	6.8	0.3202	11.4

The values of C_k and R_k have been rounded off, and R_0 was assumed to be 0.32 megohms. Experience has shown that the values for β_i (effective) can vary over 5 to 10 percent without noticeably affecting the simulator results. A 20-percent variation in neutron generation time will have little effect on the simulator. Fig.4.2-11 and 4.2-12 indicate the effect of varying l^*.

The accuracy of the simulator can be improved by increased care in the selection of the individual components, i.e., $R_6 = 11.43$ megohms and $C_6 = 6.83$ microfarads (time constant = 78.067 seconds).

The value for the capacitor C_0, which determines the mean generation time, can now be determined: $R_0 C_0 = K_{sf} l^* = 2.0 \times 10^{-5}$ seconds.

$$C_0 = \frac{10^2 \times 2.0 \times 10^{-5} \text{ seconds}}{0.320 \times 10^6 \Omega} = 0.00625 \text{ microfarads} \cong 0.006 \text{ microfarads}$$

A period computer may easily be added to the reactor simulation. Equation (1), $\phi = \phi_0 e^{t/\tau}$, is written in logarithmic form, and the derivative is taken which provides the following relationship for period:

$$\ln \phi = \ln \phi_0 + t/\tau$$

$$\frac{1}{\tau} = \frac{d}{dt} \ln \phi = \frac{1}{\phi} \frac{d\phi}{dt} \quad (42)$$

The circuit given in Figure 5-21 can be used to provide the required form. See Figure 5-22 for the relationship between period and reactivity.

The method of simulating reactor kinetics by transforming the reactor kinetic equations into a transfer function is good for reactor power range and power plant control system evaluation, but has some inherent capability limitations.

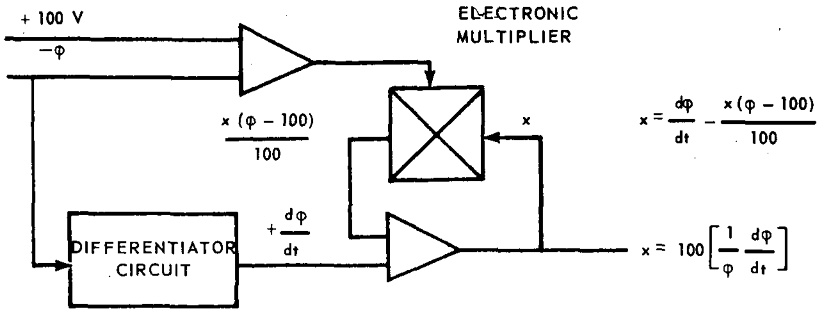


Fig. 5.21 – Circuit for computing inverse period

Satisfactory operation of an analog computer problem is difficult with signal levels of less than 1 volt because of the small signal-to-noise ratios. Since the largest practical computing voltage is 100 volts, the maximum variation in simulated reactor power that can be used is 100 to 1 or two decades. The only practical means of using the transfer function method of simulating nuclear kinetics over the entire startup range where reactor power may vary over a range as great as 10^{10} to 1, is: (1) to operate the problem until the top of a two-decade range is reached; (2) stop the problem; (3) measure the reactor power and precursor concentrations; (4) reprogram the computer with initial valves for power and precursor of 0.01 of the measured value; (5) continue the problem for the next two decades.

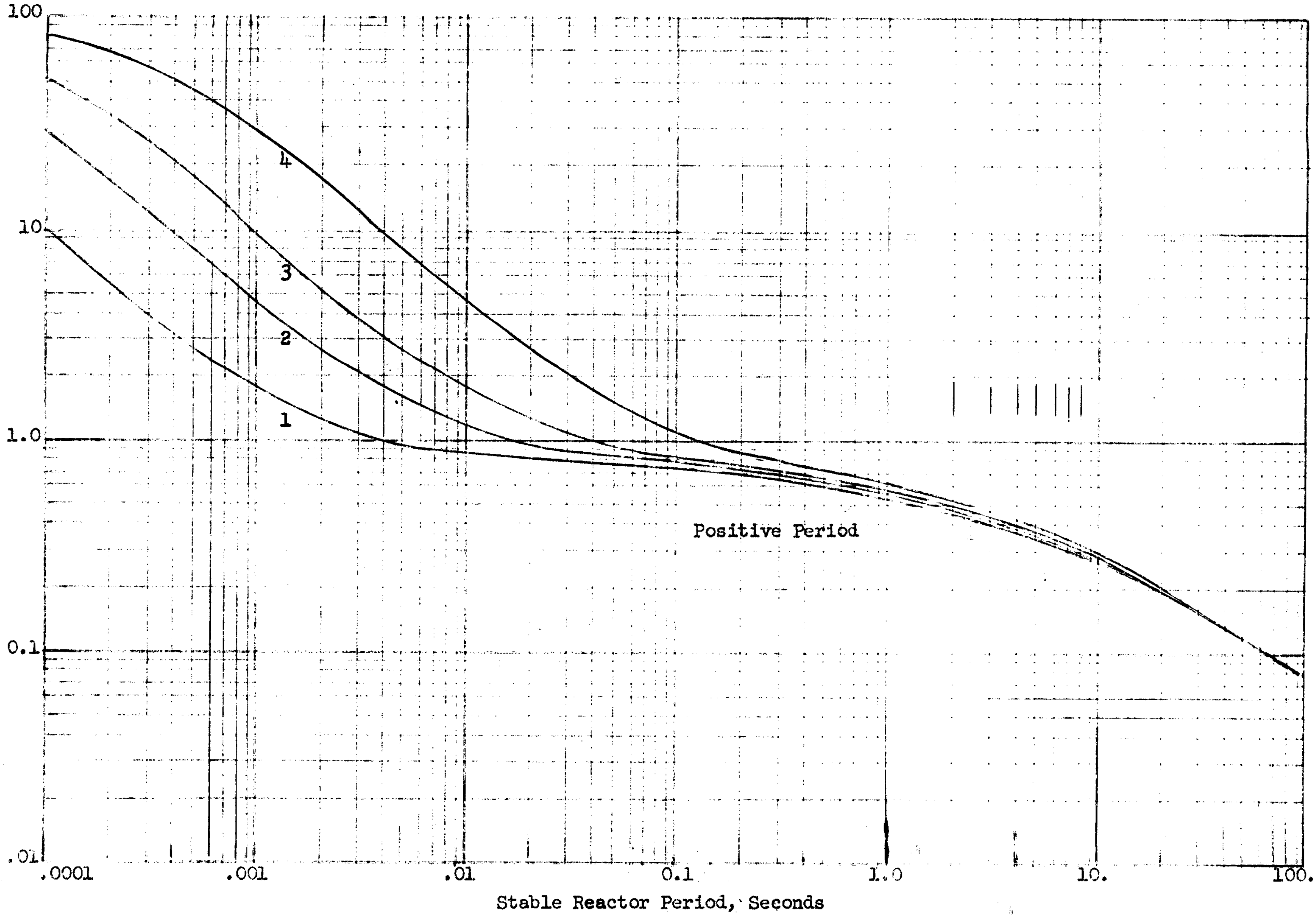
The major disadvantage of the method using transfer functions is that precursor concentration cannot be directly measured. This is not a disadvantage when a problem is started from a quiescent state, since precursor concentrations would have reached their equilibrium value for that reactor power level. When a reactor simulator is to be started, however, with the neutron flux in a transient, as in a many-decade startup problem, the concentration of precursors must be preset to their proper values, since they lag the neutron flux by as much as 80(78.125 = $1/\lambda_6$) seconds, and have a major effect on the rate of change of flux.

If the reactor equations are simulated directly, without being transformed into a transfer function, one integrator is required to solve for each of the six delayed neutron groups and another integrator is required to solve for flux. In this circuit, flux and each delayed group are available for measurement so that when a computer limit is reached, each variable is measured, divided by the desired recalibration factor (100 for 2-decade operation), and reinserted as an initial condition in each integrator. Operation can continue until another computer limit is reached. An automatic switching circuit for accomplishing this action has been developed.[11] Such a simulation is valuable in the study of reactor startup control problems, in the development of safety systems, and in providing a simulator for power plant operator tranining. Another method of simulating many decades of reactor operation uses the logarithm of nuclear power to compress the working voltage range.[12] This simulation involves a change of variables obtained by dividing the kinetic equations of the reactor by neutron flux density. A logarithmic solution is then possible that enables the many decades of reactor power to be represented on the normally 2-decade range of an analog computer. The only disadvantage in this method of simulation is that the power level output is logarithmic in form, hindering the tie-in of the simulation with the overall reactor simulation.

For a more complete simulation of an over-all propulsion system refer to APEX 800, Gas Cooled High Temperature Nuclear Reactor Design Technology, Volume 8, Control Design Factors, pp 65-100.

1. $\ell^* = 1 \times 10^{-5}$ SEC
2. $\ell^* = 4 \times 10^{-5}$ SEC
3. $\ell^* = 1 \times 10^{-4}$ SEC
4. $\ell^* = 7 \times 10^{-4}$ SEC

F[illegible]-22 - REACTIVITY VS STABLE REACTOR PERIOD



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